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Joint Nuclear Research Center
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Reactor Theory and Calculations

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Printed by L. Vanmelle, Ghent
Luxembourg, September 1970

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ABSTRACT

TIMOC is a Monte Carlo Code for the solution of the energy and time dependent (or stationary) neutron transport equation in 3-dimensional geometries. It is written in FORTRAN II and FAP and can be operated on the IBM 7090/95. The program can treat all commonly used scattering kernels, such as : isotropic and anisotropic elastic scattering, level excitation, the evaporation model, the energy transfer matrix model, which includes (n, 2n) reactions, absorption and fission. After each collision process the neutron history gets assigned a discrete energy value. The nuclear data input is, however, done by group averaged cross sections, which can be generated for any desired group structure from the ENDF/B library using the CODAC code. The geometry routines of TIMOC are exchangeable. At present the following subroutines are available : a) periodical multilayered slab-, spherical- and cylindrical lattices, b) an elaborate 3-dimensional cylindrical geometry which allows all kinds of subdivisions, c) the very flexible O5R geometry routine, which is able to describe any body or bodycombinations, with surfaces of the 2nd order.

TIMOC can generate the neutron fluxes either resulting from an external source or their fundamental mode distribution by a special source iteration procedure in eigenvalue problems. The sampling of the eigenvalues (multiplication factor, mean production and mean destruction time) is based on the life cycle point of view.

The code is equipped with several unusual variance reducing methods which can be applied optionally. In addition, a special feature allows the calculation of small perturbation effects. It is based on the method of similar flight paths and does not depend on the total variance of the considered quantity.

KEYWORDS

NEUTRON TRANSFER
MONTE CARLO METHOD
FORTRAN
FAP CODE
IBM 7090
KERNELS
MATHEMATICS

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1. INTRODUCTION *)

The necessity of performing neutron transport calculations of complicated threedimensional geometries initiated the development of the TIMOC code. Since the multigroup treatment of such systems by classical methods is beyond the capacity of contemporary computers, it was concluded that there exists a considerable incentive for the development of a Monte Carlo code. The Monte Carlo method allows the transition to a higher number of dimensions in phase space at the cost of only a linear increase in computing effort. In addition, there has accumulated some considerable experience in the formulation and application of Monte Carlo methods in connection with the FF-MOCA (1), the MOCA-2 (2) and MOCA-2A (3) code, which calculate fast effects in thermal systems. It was on the basis of these programs that in 1963 the development of TIMOC (for the IBM 7090), first described in (4), has begun.

In addition to infinite slab and cylindrical lattices, TIMOC handles spheres and an elaborate finite cylindrical geometry which allows all kinds of subdivisions. There exists also a version of TIMOC which handles the O5R geometry routine. The nuclear data input is based on the multigroup concept and most of the commonly used scattering kernels are available. It incorporates what we believe to be unusual variance reducing methods, the efficiency of which is demonstrated in (5).

The TIMOC code can be operated in several modes. They are basically distinguished by calculations in which a constant external source (in the case of stationary problems) or a δ-burst (in the case of a time de-

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*) Manuscript received on 16 June 1970

pendent study) is of interest and those in which the equilibrium or fundamental mode distribution and the corresponding eigenvalues (multiplication factor and generation time) are required. Besides of the usual reactor quantities such as flux spectra, absorption, multiplication factors etc. much emphasis is placed on the calculation of time dependent parameters. However the calculation of all these quantities is optional.

In addition, the code is equipped with a special feature allowing the estimation of small perturbation effects. This makes it possible to calculate independently these differential effects in any of the parameters computed by the code because the variance of the differential effect does not depend on the total variance of the considered quantity (reactivity, generation time, etc.).

Experience has shown that the TIMOC code can be used for complete reactor analysis and design studies (6), (7), (8) as well as for the study of more academic problems (9), (10), such as the influence of certain approximations or scattering kernels. It has also proved very useful as a provider of test results for the verification of other analystical or numerical approaches (4).

In order to facilitate nuclear data input TIMOC had been linked up with the ENDF/B cross section libraries. For this purpose the CODAC code (26) had been developed. It allows to generate group averaged nuclear parameters in the Format required by TIMOC.

2. MATHEMATICAL FORMULATION OF NEUTRON TRANSPORT IN THE MONTE CARLO THEORY

2.1 Stationary Problems

For the description of neutron transport phenomena by the Monte Carlo technique one frequently uses the integral form of the Boltzmann equation which is a Fredholm integral equation of second kind:

$$\varphi(x) = \int_R K(y \to x)\,\varphi(y)\,dy + S(x) \,. \tag{2.1}$$

For a general description of the neutron transport equation in this form we refer to the standard literature (12). Apart from an explanation of the symbols appearing in Eq.(2.1), only those quantities needed later for describing the particular Monte Carlo techniques of interest will be defined in detail.

In the case of stationary neutron transport the symbols of Eq. (2.1) have the following physical meaning:

$x = (\underline{r},\underline{E})$, the six dimensional vector in phase space, composed of the geometrical position vector $\underline{r}$ and the energy vector $\underline{E} = E\,\underline{\Omega}$, where E is the energy of the neutron and $\underline{\Omega}$ the unit direction vector ($\underline{\Omega}^2 = 1$).

$y = (\underline{r}',\underline{E}')$.

$\varphi(x)$ = Steady state neutron flux at x.

$K(y \longrightarrow x)$ = kernel describing the physical neutron transport properties, i.e. the transfer probability from point y to point x. It can be conveniently separated into two types of functions, one embodying the velocity changes and the other dealing with changes in the spatial coordinates. The velocity change part can be split into a collision kernel C and a fission kernel F:

$$K(y \rightarrow x) = \left[C(\underline{r}', \underline{E}' \rightarrow \underline{E}) + F(\underline{r}', \underline{E}' \rightarrow \underline{E})\right] T(\underline{r}' \rightarrow \underline{r}, \underline{E}).$$

The collision kernel C describes elastic and inelastic scattering by the different laws of energy transfer to be discussed later. The fission kernel F is the product of the macroscopic fission cross section Σ_f of the (incident) neutrons, the yield factor ν and the energy and direction distribution function χ of the secondary neutrons:

$$F(\underline{r}', \underline{E}' \rightarrow \underline{E}) = \nu(E') \Sigma_f(\underline{r}', E') \chi(\underline{E}).$$

The kernel T is the well known exponential degradation function for particle transport.

$S(x)$ = $\int_R d\underline{r}' S^*(\underline{r}', \underline{E}) T(\underline{r}' \rightarrow \underline{r}, \underline{E})$, which describes the uncollided neutron contribution to each phase space point x from the external source $S^*(\underline{r}, \underline{E})$.

R = Six dimensional integration space which can be split into the geometrical space B, the energy interval E_{min} to E_{max} and the velocity direction space Ω .

In the theory on which the Monte Carlo solution of Eq.(2.1) depends (13) one usually requires the conditions $S(x) > 0$ and $0 \leq \int_R K(y \rightarrow x)dx < 1$ for all $y \in R$ to be satisfied in order to obtain convergence. The first condition is satisfied for systems with an external source but not for eigenvalue calculations such as that required for a critical assembly. The second condition holds generally only for non multiplying systems. In order to get a general method which treats both fission and eigenvalue problems we split the kernel $K(y \rightarrow x)$ of Eq.(2.1) into two parts. The first, $D(y \rightarrow x)$, describes non multiplying neutron transport and the second, $P(y \rightarrow x)$, describes all neutron multiplying processes:

$$\varphi(x) = \int_R D(y \rightarrow x)\varphi(y)dy + \int_R P(y \rightarrow x)\varphi(y)dy + S(x) \qquad (2.2)$$

where

$$D(y \rightarrow x) = C(\underline{r}', \underline{E}' \rightarrow \underline{E})T(\underline{r}' \rightarrow \underline{r}, \underline{E})$$

and

$$P(y \rightarrow x) = F(\underline{r}', \underline{E}' \rightarrow \underline{E})T(\underline{r}' \rightarrow \underline{r}, \underline{E}) .$$

The method of solving Eq.(2.2) is based on the physical analogue of subsequent neutron generations. The neutrons of the initial source $S(x)$ are scattered in the system until they are lost byabsorption or leakage. In the case of fission they create the source distribution of the next cycle and so on. In mathematical terms this is the well known source iteration procedure. It can be obtained by splitting Eq.(2.2) into a system of coupled integral equations of the form:

$$\varphi_i(x) = \int_R D(y\to x)\,\varphi_i(y)\,dy + S_i(x) \tag{2.3}$$

where

$$S_1(x) = S(x) \tag{2.4}$$

and $$S_i(x) = \int_R P(y\to x)\,\varphi_{i-1}(y)\,dy\,, \qquad i=2,3,.. \tag{2.5}$$

In each equation (2.3), the following conditions are satisfied: $S_i(x) > 0$ and $0 \leqslant \int_R D(y\to x)dx < 1$ for all $y \in R$. All equations (2.3) are neutron transport equations for a non multiplying medium and can therefore be solved by ordinary Monte Carlo techniques (13).

The summation of all $\varphi_i(x)$ leads to

$$\sum_{i=1}^{N}\varphi_i(x) = \int_R D(y\to x)\sum_{i=1}^{N}\varphi_i(y)\,dy + \int_R P(y\to x)\sum_{i=1}^{N}\varphi_i(y)\,dy +$$

$$+ S(x) - \int_R P(y\to x)\,\varphi_N(y)\,dy\,. \tag{2.6}$$

As will be shown later, if the reactor is subcritical the sum $\sum_{i=1}^{\infty}\varphi_i(x) = \varphi(x)$ becomes a solution of Eq.(2.2) and $\int_R P(y\to x)\varphi_N(y)dy \to 0$ as $N\to\infty$. If, however, $\varphi_{N-1}(x)$ converges to $\varphi_N(x)$ for $N\to\infty$, i.e. if the value of $\varphi_N(x)$ does not change for higher iterations, then $\varphi_N(x)$ becomes a solution of the following homogeneous integral equation which describes a just critical system:

$$\varphi_N(x) = \int_R D(y \to x)\, \varphi_N(y)\, dy + \int_R P(y \to x)\, \varphi_N(y)\, dy. \qquad (2.7)$$

The splitting of Eq.(2.1) into neutron generations reflects the actual calculation mechanism of the Monte Carlo techniques described here. The source iteration procedure makes it easy to calculate characteristic reactor parameters and to formulate a convergence criterion for the method.

The ratio

$$k_i = \int_R S_{i+1}(x)dx \Big/ \int_R S_i(x)dx \qquad (2.8)$$

is the multiplication factor between generations "i+1" and "i" and can be obtained in this form as a sample value. When, after some cycles, the fundamental mode or equilibrium distribution is reached, then $k_{i-1} = k_i$, $\varphi_i(x) = k_i\, \varphi_{i-1}(x)$ and $\lim_{i\to\infty} k_i = k_{eff}$ becomes the parameter for stationary criticality (14,15). If $k_{eff} < 1$, Eq.(2.6) has a solution, since the sum $\sum_{i=1}^{\infty} \varphi_i(x)$ has a geometrical series as an upper limit. In the case where $k_{eff} > 1$ the procedure diverges, i.e. the calculated assembly is supercritical, but the method can also handle such problems if an artificial reduction parameter is used.

In addition to the stationary multiplication factor, defined by Eq.(2.8) the dynamical multiplication factor can also be obtained easily as a sample value. It is the average multiplication factor of all neutron generations, defined by:

$$k_{eff}(dyn) = \int_R \sum_{i=1}^{\infty} S_{i+1}(x)dx \Big/ \int_R \sum_{i=1}^{\infty} S_i(x)dx \,. \qquad (2.9)$$

2.2 Time Dependent Problems

In the study of problems of reactor kinetics, the mean neutron life-time and generation time are among the most important parameters. In order to evaluate these quantities, it is necessary to introduce the time variable explicitly into the Monte Carlo calculations.

In order to avoid confusion over the meaning of the term "neutron generation time", we recall here its basic definition (16), based on the "neutron balance" point of view. According to the elementary formulation of reactor kinetics, we have: Number of neutrons produced per unit time minus Number of Neutrons disappearing per unit time =

$$\frac{dN}{dt} = \frac{k^*_{eff} - 1}{l} N \qquad (2.10)$$

where k^*_{eff} is the neutron balance multiplication factor (16). Eq.(2.10) has the well known solution

$$N(t) = N_o \exp\left(\frac{k^*_{eff} - 1}{l} t\right) . \qquad (2.11)$$

The neutron generation time is simply the constant l appearing in these formulae. From eq.(2.11), the experimental determination of l

is achieved by observing the decay rate associated with a particular k^{*}_{eff}. In the Monte Carlo programmes described in (17) and similarly in (1) this method is used for the determination of the generation time and is called the "census taking" procedure.

Another way of defining the time eigenvalue is based on the so-called "life cycle" point of view. In this definition one refers again to the picture of successive neutron generations and defines the time eigenvalue of a multiplying system as the average time distance between two generations. It is this definition which is used in the present work owing to the great ease with which it can be extracted as a Monte Carlo sample value.

In a system which has already reached fundamental mode distribution the number of neutrons of the n'th generation is given by (16):

$$N_n = N_1 k_{eff}^{n-1} \tag{2.12}$$

If τ_{PR} is defined as the average time between two successive generations, the time between the first and n'th generation is $t = (n-1)\tau_{PR}$. Substituting this into Eq.(2.12) gives

$$N(t) = N_1 k_{eff}^{t/\tau_{PR}} \tag{2.13}$$

or

$$\frac{1}{N}\frac{dN}{dt} = \frac{\ln k_{eff}}{\tau_{PR}} \tag{2.14}$$

and if k_{eff} is close to unity Eq.(2.14) can be approximated by:

$$\frac{dN}{dt} = \frac{k_{eff} - 1}{\tau_{PR}} N \qquad (2.15)$$

which is identical with equation (2.10) and implies that $\tau_{PR} = l$ and $k_{eff} = k^*_{eff}$. As will be shown later, the Monte Carlo procedure provides a direct and very exact estimate of the quantity τ_{PR}.

It should be pointed out clearly here that the definitions of both k_{eff} and k^*_{eff} (life cycle and neutron balance) and therefore also of τ_{PR} and l are different. Only in the case of a nearly critical system ($\sim \pm$ 4% from criticality) the numerical values of the quantities obtained by the life cycle point of view and the neutron balance point of view coincide. In systems where the multiplication rate differs by a larger amount from criticality, the only invariant quantity is the decay constant α, which in neutron balance calculations is given by:

$$\alpha = \frac{k^*_{eff} - 1}{l} \qquad (2.16)$$

and in life cycle calculation by:

$$\alpha = \frac{\ln k_{eff}}{\tau_{PR}} \; . \qquad (2.17)$$

These considerations will become important when we compare numerical values obtained by different methods.

In order to calculate the quantity τ_{PR}, i.e. the mean time between the (i-1)'th and i'th generation, it is assumed that at t = 0 a neutron population of the energy and space distribution of the (i-1)'th gene-

ration is inserted into the assembly. Only the die-away of this population is of interest and the estimates of the time dependent parameters are therefore carried out for this one neutron generation. The relevant form of the transport equation is:

$$\varphi_i(x,t) = \int_R dy \int_0^t dt' \delta(t-t'-\frac{|r-r'|}{\sqrt{E}}) D(y\rightarrow x)\varphi_i(y,t') + \\ + S_i(x)\delta(t). \qquad (2.18)$$

The solution of Eq.(2.18) is equivalent to the solution of Eq.(2.3) with the only difference that the time variable is added to the phase space. In physical terms Eq.(2.18) means that a neutron population with a distribution $S_i(x)$ is inserted into the reactor at t = 0. It has the solution $\varphi_i(x,t)$, which means that sample values of each neutron history are taken as a function of the time from the moment of insertion.

For simplicity we shall assume in the following that the i-th neutron generation always represents the fundamental mode distribution so that the index "i" can be dropped from all quantities except $\varphi_i(x,t)$, where it indicates that just one generation is considered.

Two sample values which can be obtained from the solution $\varphi_i(x,t)$ will be of particular interest. One is the time dependent build up of the source neutrons $S_{i+1}(x,t)$ of the next generation, which we call here $N_p(t)$. The other is the time dependent neutron loss of the generation under consideration described in the following by $N_D(t)$. If $p^*(x)$ is the appropriate production kernel and $D^*(x)$ the absorption

and leakage kernel, i.e. $P^*(y) = \int_R P(y \to x)dx$ and $D^*(y) = 1 - \int_R D(y \to x)dx$ then we can write:

$$N_{PR}(t) = \int_R P^*(x)\varphi_i(x,t)dx \tag{2.19a}$$

or:

$$N_{PR}(t) = \iint_R dr\,dE\,\nu(E)\Sigma_f(r,E)\varphi_i(r,E,t). \tag{2.19b}$$

Similarly, if $D^*(x)$ is the absorption and leakage kernel, then the destruction rate is given by

$$N_{DE}(t) = \int_R D^*(x)\varphi_i(x,t)dx \tag{2.20a}$$

or:

$$N_{DE}(t) = \iint_R dr\,dE\,\Sigma_a(r,E)\varphi_i(r,E,t) + \int_B \mathrm{grad}_n\,\varphi_i(r,E,t)ds \tag{2.20b}$$

The first integral on the right hand side stands for the time dependent neutron loss by absorption (Σ_a is the macroscopic absorption cross section) and the surface integral for the time dependent neutron loss by leakage through the outer boundary, n being the normal to the surface and ds the surface element.

By the usual first moment calculations, the time averages of $N_{PR}(t)$ and $N_{DE}(t)$ can now be evaluated.

The first time moment of Eq.(2.19) has the form:

$$\tau_{PR} = \int_R dx \int_0^\infty t\,dt\, P^*(x)\,\varphi_i(x,t) \Big/ \int_R dx \int_0^\infty dt\, P^*(x)\,\varphi_i(x,t) \qquad (2.21)$$

where

$$\int_R dx \int_0^\infty dt\, P^*(x)\,\varphi_i(x,t) = k_{eff} \int_R S_i(x)\,dx \qquad (2.22)$$

if one makes use of the relations (2.5) and (2.8). The value τ_p can be interpreted as the mean time necessary to build up the next following neutron generation. It corresponds therefore to the generation time definition of the "life cycle" point of view.

The time average of Eq. (2.20) has to be interpreted as the mean time a neutron stays in the assembly under consideration. It is defined as:

$$\tau_{DE} = \int_R dx \int_0^\infty t\,dt\, D^*(x)\,\varphi_i(x,t) \Big/ \int_R S_i(x)\,dx \qquad (2.23)$$

where, for physical reasons:

$$\int_R S_i(x)\,dx = \int_R dx \int_0^\infty dt\, D^*(x)\,\varphi_i(x,t). \qquad (2.24)$$

It can be seen that there is generally no relation between τ_p and τ_D unless

$$\int_R \left[P^*(x) - k_{eff}\, D^*(x)\right] \varphi_i(x,t)\,dx = 0 \qquad \text{for } \varphi_i(x,t) \neq 0. \qquad (2.25)$$

In this case $\tau_{PR} = \tau_{DE}$. This condition is only satisfied for certain calculational models or material compositions.

It is worth mentioning that in the usual "neutron balance" method of calculating a time eigenvalue, the generation time is defined by a relation using the concept of "importance":

$$\ell = \frac{\iint_R \frac{1}{\sqrt{E}} \varphi^+(\underline{r},\underline{E})\varphi(\underline{r},\underline{E})\,d\underline{r}\,d\underline{E}}{\iint_R \varphi^+(\underline{r},\underline{E})\chi(\underline{E})\,d\underline{r}\,d\underline{E}\iint_R \nu(\underline{E}')\Sigma_f(\underline{r}',\underline{E}')\varphi(\underline{r}',\underline{E}')\,d\underline{r}'\,d\underline{E}'} \qquad (2.26)$$

where the notation of Usachoff (14) has been used. The importance function or adjoint flux is denoted by $\varphi^+(\underline{r},\underline{E})$ in this expression. The quantity corresponding to ℓ from the "life cycle" viewpoint is the parameter τ_{PR}, while the quantity τ_{DE} is matched in the neutron balance framework by a parameter ℓ' obtained by setting $\varphi^+(\underline{r},\underline{E}) = 1$ in Eq.(2.26), i.e.:

$$\ell' = \frac{\iint_R \frac{1}{\sqrt{E}} \varphi(\underline{r},\underline{E})\,d\underline{r}\,d\underline{E}}{\iint_R \nu(E)\Sigma_f(\underline{r},E)\varphi(\underline{r},\underline{E})\,d\underline{r}\,d\underline{E}} \qquad (2.27)$$

Only for one velocity, infinite medium calculations do the quantities ℓ and ℓ' become equal, since in this case φ^+ is constant. It can be shown easily in this case that Eq.(2.25) is also satisfied, with the consequence that τ_{PR} becomes equal to τ_{DE}.

The difference between τ_{PR} and τ_{DE} can be understood in physical terms from the fact that there exists no direct relation between the time a neutron stays in a system (for instance core and reflector) and

the time it needs for its reproduction. For physical reasons it is also clear that $\tau_{DE} \geq \tau_{PR}$. If $\tau_{PR} = \tau_{DE}$, the lowest possible neutron population in a critical system is theoretically one neutron. If however $\tau_{DE} > \tau_{PR}$ the minimum number of neutrons becomes τ_{DE} / τ_{PR}.

3. LIST OF SYMBOLS

In the following chapters a large number of symbols are used. In order to maintain a certain consistency and reduce confusion, some rules have been established. In an effort to be consistent with conventional terminology each symbol is, with very few obvious exceptions, assigned only one meaning.

The rules are as follows:

a) Single roman letters from g to n are used as counting indices only. The corresponding capital letters are the maximum indices to which counting is done. These letters refer to geometrical regions, isotope mixtures and energy groups. In the symbolic flow charts the greek letters α to δ are also used as counting indices.

b) Single letters, apart from thosementioned above, and all greek letters describe variables and functions.

c) Double capital letters used as a subscript or superscript describe a nuclear reaction type or a sampling procedure, e.g. FI = fission, PR = production, RR = Russian Roulette, etc.

LIST OF SYMBOLS

Symbol	Meaning
A(k)	mass number of isotope k
AB	index for absorption
B	free variable
C	C = 0.6745
BC	boundary crossing
CA	index for capture
CL	index for collision
D	neutron track length (e.g. distance between two successive collision points) [cm]
DE	index for neutron destruction (e.g. absorption, leakage etc.)
E	energy variable
ED	index for energy deposition
EL	index for elastic scattering
F	inelastic multiplication factor (e.g. in case of (n, 2n) reactions)
F	surface of region
g, G	number of Russian Roulette games in a history or free counting index for partial cross sections like excited levels etc.
h, H	counting variable for primary neutron histories
i, I	counting variable for energy groups
IN	index for inelastic scattering
IT	index for the ITER*Version (eigenvalue calculations)
j, J	counting variable for collisions; usually a function of h or n, i.e.: j = h(j) or n(j)

k, K	isotope identification variable
k_{eff}	multiplication factor,
l, L	identification variable for geometrical regions
LE	index for leakage
LV	index for excited level
m, M	identification variable for the isotope mixtures
max	maximum value of a variable
n, N	counting variable for secondary neutron histories
P	probability variable; $0 \leqslant P \leqslant 1$
PR	index for neutron production by fission
Q	normalized sample value of a certain reaction rate (e.g. Q_{AB} = mean number of absorptions per initial neutron in a system)
r	space vector
R	distance from a collision point to the outer surface along the neutron flight vector
RR	index for Russian Roulette
s, S	normalization variables
SC	index for scattering
SD	index for slowing down (e.g.: W^{SD} neutron weight after slowing down below the cut off energy)
t	time variable
T	nuclear temperature
TO	index for total cross section
V	neutron velocity $\left[\text{cm sec}^{-1}\right]$
Var	variance
W	neutron weight
x, y, z	space coordinates

α, A	counting index
α_g	coefficients of the polynomial describing the energy dependence of ν (mean number of neutrons per fission)
α_1, α_2	coefficients in the formula for minimum variance combination of two sample values
β, B γ, Γ δ, Δ	) counting index
ϑ, Θ	azimuthal angle
λ	neutron flight length in mean free paths
Λ	modified flight length in the ELP sampling procedure
$\bar{\mu}$	average cosine of the scattering angle (lab. system)
ν	mean number of neutrons per fission
ξ	random number equally distributed in (0,1)
ϱ	normalization factor if Russian Roulett, otherwise $\varrho = 1.0$
σ	microscopic cross section
Σ	macroscopic cross section
τ	mean times (e.g. τ_{PR} = mean production time) or Fermi Age
φ, ϕ	horizontal angle
$\chi(E)$	energy distribution function of the fission neutrons
ψ	fractional generated neutrons
Ω	direction of the neutron path

4. VARIANCE REDUCING METHODS AND SAMPLING TECHNIQUES

As already mentioned in the introduction a number of variance reducing methods were incorporated into the TIMOC code. The basic idea of the different methods is described in the following and as far as possible reference is made to the corresponding literature. For simplicity all considerations are limited in this section to one geometrical region and one energy group. The extension to more regions and energy groups does not essentially change the methods. The method of Expected Values characterized by the introduction of weight factors, usually applied to decrease the variance of the absorption probability, was extended to obtain better estimates also for the fission and leakage probabilities.

4.1 The "Standard" Weight Estimator

Instead of treating absorption as a random event, one assigns to each neutron history at its origin (t = 0) a weight factor W_o. At the j'th collision point a time distance t has been reached from the origin and the weight $W_{h,j}(t)$ (h is the history index) has the value

$$W_{h,j}(t) = W_o \left(\prod_{g=1}^{j} P_{sc} \right) \delta \left(t - \sum_{g=1}^{j} \frac{D_{h,g}}{\sqrt{E}} \right) \qquad (4.1)$$

where $D_{h,j}$ is the geometrical travel distance between the collision point g-1 and g. $D_{h,g}$ is picked from an exponential distribution, and is analytically given by

$$D_{h,g} = -\frac{1}{\Sigma_{TO}} \ln(1-\xi) \qquad (4.2)$$

and

$$P_{SC} = \Sigma_{SC} / \Sigma_{TO}$$

where ξ is a random number uniformly distributed in (0,1). Σ_{TO} is the macroscopic total cross section, Σ_{SC} the macroscopic scattering cross section, and $\sqrt{E}$ the neutron velocity (apart from a constant).

Since frequent reference is made to this function we call it in this paper the "Standard"Weight Estimator". As will be shown later, several sample values can be obtained from it. They are referred to as values obtained by the "Standard Version" of the TIMOC code.

4.2 The Russian Roulette Game

The introduction of the "Standard Weight Estimator" as described above requires in many problems that histories are truncated to avoid calculations with parts of histories which have small weights and do not therefore contribute significantly to the results, though they consume large amounts of computer time. For this reason the sampling procedure of the Russian Roulette game is introduced and is shown in a simplified flow diagram in Fig. 1. At the beginningof each history the weight W is set to unity (W_o = 1.0). At each collision point the weight W_j is multiplied

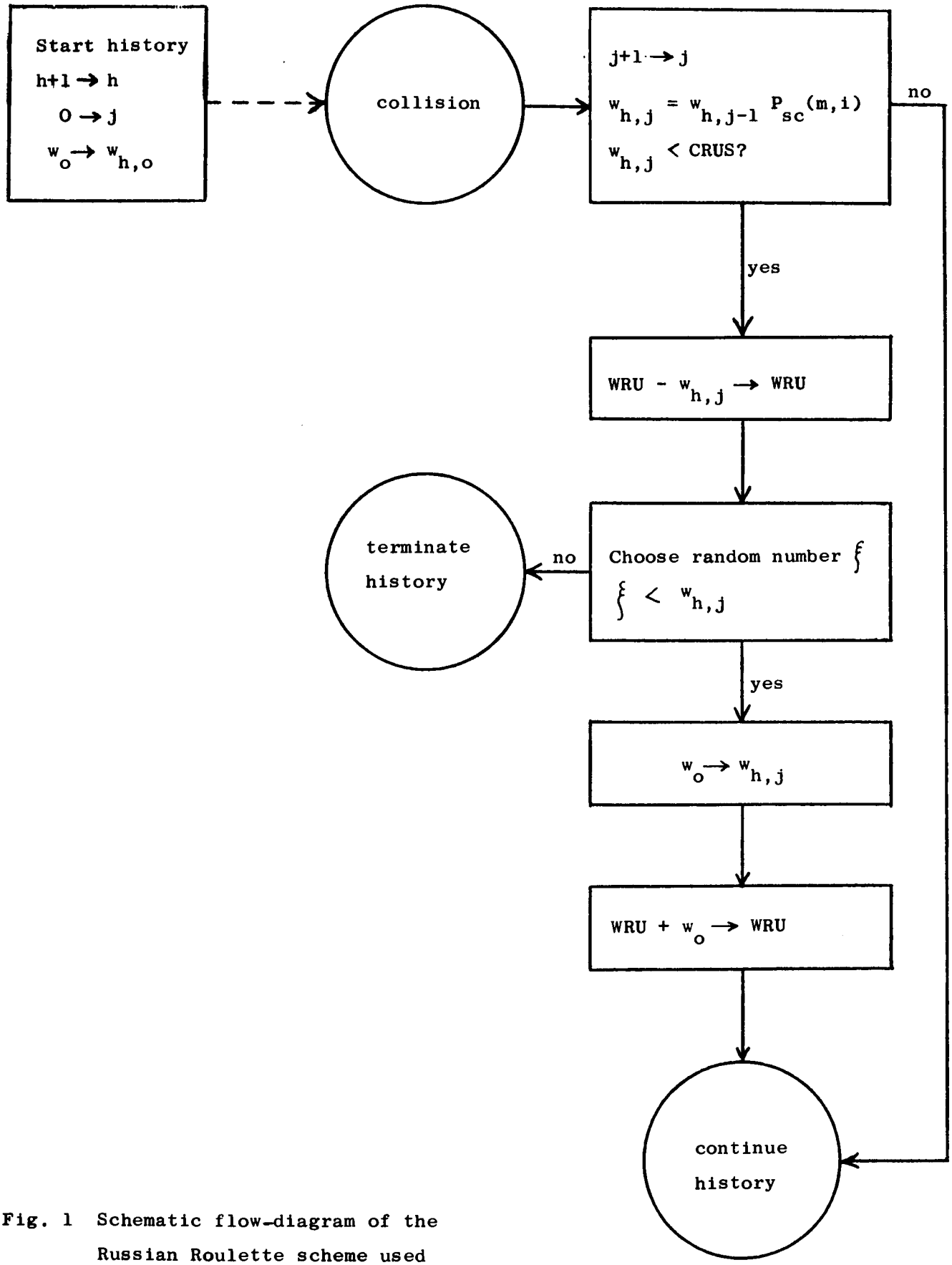


Fig. 1 Schematic flow-diagram of the Russian Roulette scheme used in TIMOC

by the scattering probability $P_{sc} \leqslant 1.0$ and afterwards a test is made to see whether the weight is less than some constant CRUS. If not, the history is continued; if so, the Russian roulette game is played by chosing a random number equally distributed in (0,1). If this is smaller than the weight, the weight is reset to unity (W_o) and the history is continued; otherwise the current history is terminated and a new one started. This procedure means that the weight is reset to unity with a probability W and to zero (termination of the history) with a probability (1-W), so that the averaged weight continued is W.

Since this procedure is a statistical process, it has a certain error margin and one will find at the end of a calculation that the average number of neutrons lost, for instance by absorption and leakage, is not exactly the same as the number of neutrons introduced into the system. One can get rid of this inconvenience by a modified normalizing procedure where the final results are multiplied by a correction factor ρ. Instead of only dividing by the sum of the initial weight HW_o (where H is the total number of calculated histories) one uses a corrected quantity where the actual weights of the Russian roulette procedure are taken into account. This quantity is $\sum_h (G_h W_{h,o} - \sum_{g(h)} W_{h,g}^{RR})$ where G_h is the number of times the weight factor W is reset to unity during the course of history h. The quantity $W_{h,g}^{RR}$ is the weight factor $W_{h,j}$ at the moment the Russian Roulette game is applied (i.e. if the weight of history h is ei er reset to $W_{h,o}$ or terminated). It can be shown that the sum $\sum_h \sum_g W_{h,g}^{RR}$ approaches on the average $\sum G_h W_{h,o}$.

The normalization factor takes the form:

$$\sum_h W_{h,o} + \sum_h \left(G_h W_{h,o} - \sum_g W_{h,g}^{RR} \right) = \sum_h \sum_g \left(W_{h,o} - W_{h,g}^{RR} \right)$$

where

$$\sum_{g=o}^{o} \left(W_{h,o} - W_{h,g}^{RR} \right) = W_{h,o} - W_{h,o}^{RR} = W_{h,o} - W_{h,J}$$

and

$$\sum_{g=o}^{1} \left(W_{h,o} - W_{h,g}^{RR} \right) = 2W_{h,o} - W_{h,o}^{RR} - W_{h,1}^{RR} .$$

The correction factor ϱ reads then

$$\varrho = HW_o \left[\sum_h \sum_g \left(W_{h,o} - W_{h,g}^{RR} \right) . \right. \tag{4.3}$$

The introduction of this correction factor ϱ looks rather arbitrary and even doubtful. A justification can however be found in the fact that one can either normalize by the number of neutrons started during a calculation or the total number of neutrons lost (by leakage, absorption etc.). Naturally these two quantities have to be identical except if Russian Roulette is played, where they show statistical deviations. A correction by the factor ϱ is therefore nothing more than a normalization by the number of neutrons lost. In calculations of non multiplying media and in eigenvalue calculations it results in the fact that all losses add up to unity.

As shown by Rotenberg (19) some attention has to be given to finding the optimum value of CRUS. In (19) it is stated: "Theoretically it should be 1.0 since then at each collision a large weight is being processed. Actually the cost is slightly decreased when CRUS is in the range $0.4 \leq \text{CRUS} \leq 0.75$ (depending on the problem to be solved) because the time taken to initialize and terminate a history is not negligible and also because, although a small CRUS causes more time to be spent in processing parts of histories with small weights, it results in a smaller variance".

4.3 The Method of Fractional Generated Neutrons

During the die away of the neutrons of one generation, new neutrons of the next generation are born. In the TIMOC code these are treated by the Method of Fractional Generated Neutrons. Fractional neutrons are generated at each collision point and for each fissionable isotope k as a function of time. They are then accumulated for each isotope separately until their sum reaches the magnitude of the initial weight factor, W_o, or (depending on the option) the value of k_{eff} which is periodically estimated during the calculation. Thus, the weight $\psi_{h^*,j}(k)$ which is generated up to the h^*-th history, can be expressed by

$$\psi_{h^*,j}(k) = \left\{ \sum_{h=1}^{h^*} \sum_{g=1}^{j} P_{PR}(k) W_{h,g-1} \right\} \quad \text{mod } (W_o \text{ or } k_{eff}) \qquad (4.4)$$

where $P_{PR}(k) = \nu(k) \Sigma_{FI}(k) / \Sigma_{TO}$ and k is the isotope index.

For further information see (2). When $\psi_{h^*,j}(k)$ becomes equal to or greater than W_o or the estimate of k_{eff}, the neutron position and other characteristic parameters (such as the time etc.) are stored in a buffer storage and $\psi_{h^*,j}(k)$ is reduced by W_o or k_{eff}. The current history is then continued normally and the accumulated supply of fractional neutrons identified with a new neutron. The history of this neutron, however, belongs to the next generation and is started according to the previously stored information. In the present version of the code, the buffer storage accepted the characteristic parameters of 400 secondary neutrons.

4.4 The Method of Expected Leakage probability

This method may be used as an optional version in the TIMOC code. For simplicity we describe here the principles of the ELP method for monoenergetic neutrons in a homogeneous one region sphere. The extension to more complicated geometries and energy dependent scattering kernels is rather obvious.

In the Analogue Monte Carlo technique a neutron history would be followed from collision to collision until the particle leaks out of the assembly, at which point the history is finished. The time dependent neutron distribution function and leakage rate are obtained by taking samples at appropriate points along these flight paths.

In the method of Expected Leakage Probability this random process, which is a direct picture of nature, is changed. Instead of thinking of the

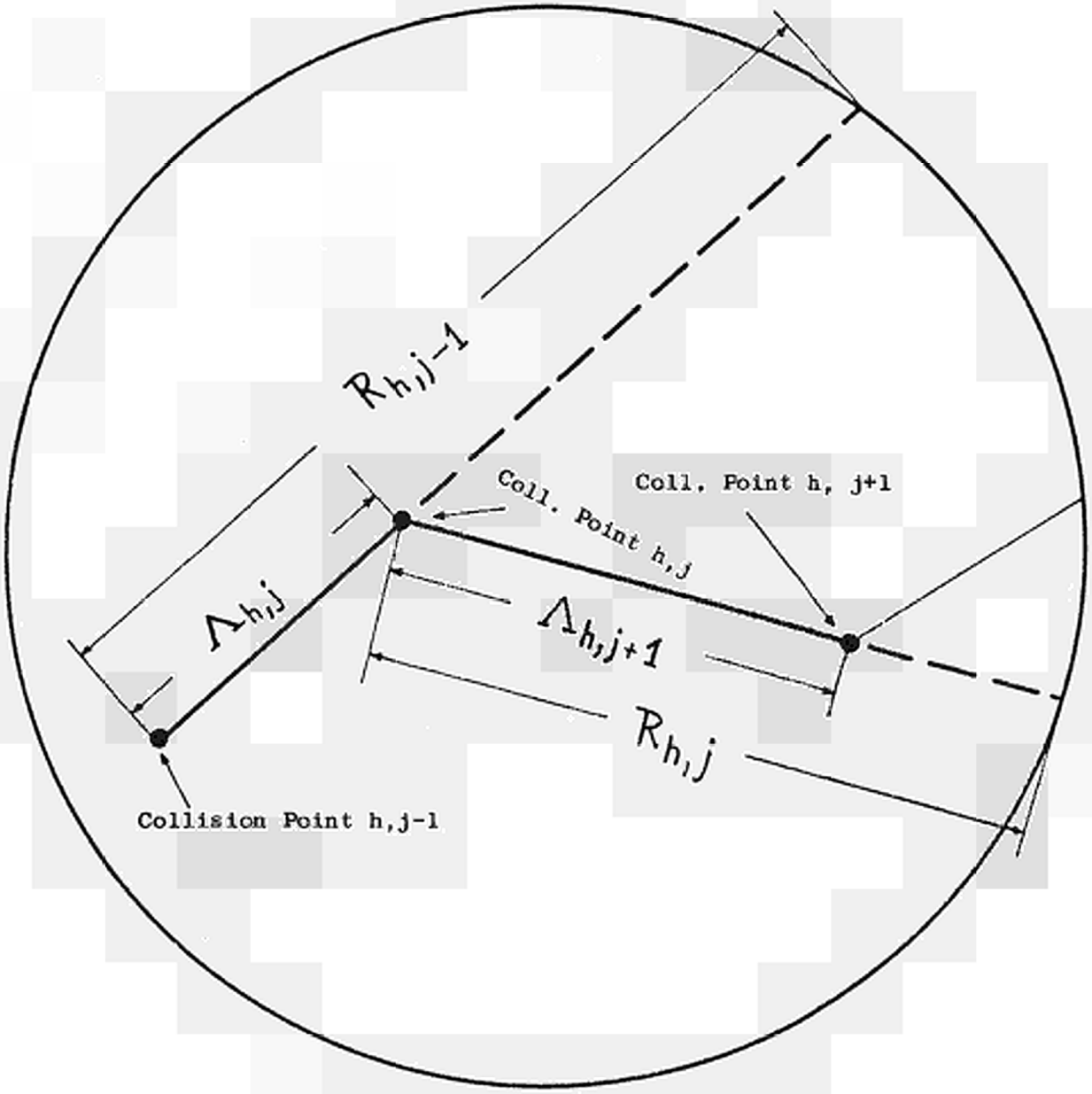


Fig. 2 A typical collision scheme of a neutron batch in the ELP method including the appropriate flight distances.

neutrons as they really exist, we group them into hypothetical "batches". The behaviour of the neutron population is then described in terms of the history of these batches. To each "batch" we assign a weight factor W, a time t' and a direction vector $\vec{\Omega}$ when the batch starts or emerges from a collision point (see Fig. 2). The weight factor W is split into two parts corresponding on the one hand to the proability P_{SC} that the batch makes its next collision within the sphere along the direction vector $R\vec{\Omega}$ and on the other hand to the probability P_{LE} that the batch escapes from the sphere.

Using the well known exponential degradation function

$$P(x)dx = \Sigma_{TO} \exp(-\Sigma_{TO} x)dx, \quad 0<x\leqslant R, \tag{4.5}$$

one obtains for the leakage proability

$$P_{LE} = 1 - P_{SC} = 1 - \int_0^R P(x)dx = \exp(-\Sigma_{TO} R) \tag{4.6}$$

where Σ_{TO} is the total macroscopic cross section of the sphere and R the distance from the starting or collision point to the outer boundary of the sphere along the flight vector $\vec{\Omega}$.

From every collision or starting point, it is supposed that a weight $P_{LE}\ W = W^{LE}$ leaks out of the system while a part $P_{SC}\ W$ is retained, corresponding to the occurrence of a further collision somewhere on the flight path $R\vec{\Omega}$. For the latter, retained, part of the neutron batch there exists the following <u>normalized</u> exponential distribution

$$P^*(x)dx = \frac{\Sigma_{TO}}{P_{SC}} \exp(-\Sigma_{TO}x)dx, \quad 0<x\leqslant R. \tag{4.7}$$

The migration distance for the random walk process of the neutron batch remaining in the assembly is therefore given by

$$\Lambda = -\frac{1}{\Sigma_{TO}} \ln(1-P_s\xi) \tag{4.8}$$

where ξ is a random number uniformely distributed in (0,1).

Applying, in addition to the standard weight estimator, the "Method of Expected Leakage Probability", the time dependent weight factor can be expressed by

$$W^*_{h,i}(t) = W_o\left\{\prod_{j=1}^{i-1} P_{SC}\left[1-\exp\left(-\Sigma_{TO}R_{h,j-1}\right)\right]\right\}\cdot$$

$$\cdot\left[1-\exp\left(-\Sigma_{TO}R_{h,i-1}\right)\right]\delta\left(t-\sum_{j=1}^{i}\frac{\Lambda_{h,j}}{\sqrt{E}}\right) \tag{4.9}$$

where $R_{h,j}$ is the distance between collision point j and outer boundary of the assembly along the direction of the neutron flight path, $\exp(-\Sigma_{TO}\cdot R_{h,j})$ the corresponding leakage probability P_{LE} and $\Lambda_{h,j}$ the distance between collision point j-1 and j calculated by Eq. (4.8). The function $W^*_{h,i}(t)$ is called the "Expected Leakage" Estimator. Sample values obtained by this weight estimator will be referred to later on as the results from the Expected Leakage Probability (or ELP) version.

The application of this technique (ELP) has the result that a neutron history is never finished. However, in practical calculations one is only interested in the behaviour of the population within a certain finite time range. Thus it is quite obvious that all neutron histories must be terminated when their total duration t satisfies

$$t > t_{max}$$

where t_{max} is the maximum time of interest in any particular problem.

The description of a neutron population on the basis of the above scheme permits the sampling of any quantities of interest whatever. The time dependent leakage, collision density, absorption etc., all are equally accessible. Since the time dependent leakage $Q_{LE}(t)$ is of particular interest for comparisons with experimental measurements, the sampling formula for this quantity will be given explicitly:

$$Q_{LE}(t) = \frac{\varrho}{H} \sum_{h=1}^{H} \left\{ \prod_{j=1}^{J} (1 - P_{h,j}^{LE}) \right\} P_{h,J+1}^{LE} \cdot$$

$$\cdot \delta\left(t - \sum_{j=1}^{J} \frac{D_{h,j}}{\sqrt{E}} - \frac{R_{h,J+1}}{\sqrt{E}}\right), \quad 0 < t \leqslant t_{max} \qquad (4.10)$$

where $P_{h,j}^{Le}$ is the leakage probability after the (j-1)'th collision in the h'th neutron history. The δ function shows that the time t is fixed by the path lengths $\sum_{j=1}^{J} D_{h,j} + R_{h,J+1}$ and the velocity $\sqrt{E}$.

With regard to the calculation of time dependent leakage, one can already see from the above discussion that the ELP is more powerful than the AMC method for the following two qualitative reasons:

Each neutron history can be carried on for any desired duration in time, in contrast to the AMC method where a history is finished when the particle leaks out of the assembly.

At each collision point one gets a contribution to the leakage, whereas, in the AMC method each neutron history contributes only once to the leakage.

On the other hand, some additional computing effort is required in the ELP method because of the need to evaluate an exponential which is not present in the AMC. The extra computing time involved is not, however, important for the cases of interest.

4.5 Semi-Systematic Sampling

This method is a general and powerful variance reducing method which can be applied where tables of equal probability intervals are used. A general description of the method has been published by Steinberg (20). The basic idea of the method consists in choosing a pseudo random process which ensures that during the course of a calculation each probability interval is taken with the same frequency. In other words, a certain probability interval of the same variable can only be used a second time if all other intervals were already used once and so on, which results in a random permutation of the intervals. In the TIMOC code, semi-systematic sampling is used, for example, in the determination of the exponentially distributed neutron flight path and the starting energy of primary and secondary neutrons.

4.6 Double Sampling for Multiplication Factor and Absorption Calculations

For the calculation of the multiplication factor and the absorption, two nearly uncorrelated sampling procedures are used and their results are combined by a procedure which minimizes the statistical errors. Since the multiplication factor and total absorption are time independent quantities, we shall drop the time variable for convenience. This is equivalent to an integration over the considered time interval.

One sampling method follows the usual technique of calculating the absorption and fission probability at each collision point and normalizing the sum. Thus:

$$k_{eff}^{coll} = \sum_{h=1}^{H} \left(\sum_{j=1}^{J} W_{h,j-1} P_{h,j-1}^{PR} \right) \Big/ HW_o \qquad (4.11)$$

and

$$A^{coll} = \sum_{h=1}^{H} \left(\sum_{j=1}^{J} W_{h,j-1} P_{h,j-1}^{AB} \right) \Big/ HW_o \qquad (4.12)$$

where H is the total number of histories handled and j the number of collisions a neutron makes during its history. $W_{h,j}$ is the "Standard Weight Estimator" (4.1) or the "Expected Leakage Weight Estimator" (4.9).

The second sampling method makes use of the flux calculations which are carried out simultaneously with the course of a history. At each collision point (or boundary crossing) the actual neutron track length is calculated. This length is multiplied by the current weight of the neutron and the product is recorded as a function of the geometrical region and energy group.

Thus denoting the track length between two collisions by D_j, the sums $\sum_{j=1}^{J} D_j W_{j-1}$ over all collisions are collected and the contribution from the track length between the last collision and the outer boundary $R_J W_J$ is added. After dividing this sum by the number of histories one obtains the volume integrated flux. This method is in some cases more powerful than the usual one where the neutron flux is calculated from the collision density. It becomes evident if one thinks of void regions in a reactor. According to the above prescription, one obtains for the "Standard Weight Estimator" (4.1) the following expression for the flux:

$$\phi = \sum_{h=1}^{H} \left[\left(\sum_{j=1}^{J} D_{h,j} W_{h,j-1} \right) + R_{h,J} W_{h,J} \right] \Big/ V_l H W_o \qquad (4.13)$$

where V_l is the volume of space region l .

If the "Expected Leakage Estimator" is used, two contributions to the flux have to be considered. One is the path length D_j between the collision points and the second is the distance R_j between the collision point and the outer boundary of the assembly (see Fig. 2). In this case the flux estimator becomes

$$\phi^* = \sum_{h=1}^{H} \left(\sum_{j=1}^{J} D_{h,j} W^*_{h,j} + R_{h,j-1} W^{*LE}_{h,j} \right) \Big/ V_l H W_o \qquad (4.14)$$

where

$$W^*_{h,j} = W_o \left\{ \prod_{g=1}^{j-1} P_{sc} \left[1 - \exp\left(- \Sigma_{TO} R_{h,g-1} \right) \right] \right\} \cdot \qquad (4.15)$$

$$\cdot \left[1 - \exp\left(- \Sigma_{TO} R_{h,j-1} \right) \right]$$

and

$$W_{h,j}^{*LE} = W_o \left\{ \prod_{g=1}^{j-1} P_{sc} \left[1 - \exp\left(- \sum_{TO} R_{h,g-1} \right) \right] \right\} \exp\left(- \sum_{TO} R_{h,j-1} \right). \tag{4.16}$$

All reaction rates are obtained by multiplying the flux value by the corresponding macroscopic cross section. This leads, for instance in the case where the fission rate or the absorption is calculated, to a sample value which is independent of the one calculated by Eqs. (4.11) and (4.12). Using the "Standard Weight Estimator" this second sample value for the multiplication factor is obtained by the relation:

$$k_{eff}^{flux} = \left[\sum_{h=1}^{H} \left(\sum_{j=1}^{J} D_{h,j} W_{h,j-1} \nu \sum_{FI} \right) + R_{h,J} W_{h,J} \nu \sum_{FI} \right] \Big/ H W_o \tag{4.17}$$

and for the absorption:

$$A^{flux} = \left[\sum_{h=1}^{H} \left(\sum_{j=1}^{J} D_{h,j} W_{h,j-1} \sum_{AB} \right) + R_{h,J} W_{h,J} \sum_{AB} \right] \Big/ H W_o . \tag{4.18}$$

It can be calculated at almost no extra computer cost, since all the parameters which are needed have to be calculated anyway. Results show, however, that by the combination of both estimates one can obtain a decrease in the probable error and a significant improvement in the results.

As is known from the theory of statistics, the two sample values of the same quantity (for instance k_{eff}) can be combined in an optimum manner if one knows the variance and covariance of both samples. For this purpose one has to calculate at <u>the end</u> of each history the quantities

$$k_h^{coll} = \sum_{j=1}^{J} W_{h,j-1} P_{PR} \tag{4.19}$$

and

$$k_h^{flux} = \left(\sum_{j=1}^{J} D_{h,j} W_{h,j-1} \nu \Sigma_{FI}\right) + R_{h,J} W_{h,J} \nu \Sigma_{FI} \tag{4.20}$$

and to combine the squares in the following manner

$$Var(k_{eff}^{coll}) = \left[\sum_{h=1}^{H} (k_h^{coll})^2 / HW_o^2\right] - (k_{eff}^{coll})^2, \tag{4.21}$$

$$Var(k_{eff}^{flux}) = \left[\sum_{h=1}^{H} (k_h^{flux})^2 / HW_o^2\right] - (k_{eff}^{flux})^2. \tag{4.22}$$

In the same manner one can calculate at the end of each history the correlation factor from which the covariance can be calculated:

$$Cov(k_{eff}^{coll}, k_{eff}^{flux}) = \left[\sum_{h=1}^{H} (k_h^{coll} k_h^{flux}) / HW_o^2\right] - k_{eff}^{coll} k_{eff}^{flux}. \tag{4.23}$$

The averaging of the two estimates of k_{eff} is then peformed by the usual procedure of minimizing the variance of the following combination (21):

$$k_{eff} = \alpha_1 k_{eff}^{coll} + \alpha_2 k_{eff}^{flux}. \tag{4.24}$$

This leads to

$$\alpha_1 = \frac{1}{B}\left[\mathrm{Var}(k_{eff}^{flux}) - \mathrm{Cov}(k_{eff}^{coll}, k_{eff}^{flux})\right], \tag{4.25}$$

$$\alpha_2 = \frac{1}{B}\left[\mathrm{Var}(k_{eff}^{coll}) - \mathrm{Cov}(k_{eff}^{coll}, k_{eff}^{flux})\right] \tag{4.26}$$

where

$$B = \mathrm{Var}(k_{eff}^{coll}) + \mathrm{Var}(k_{eff}^{flux}) - 2\,\mathrm{Cov}(k_{eff}^{coll}, k_{eff}^{flux}). \tag{4.27}$$

The variance of the averaged k_{eff} is then given by:

$$\mathrm{Var}(k_{eff}) = \frac{1}{B}\left[\mathrm{Var}(k_{eff}^{coll})\mathrm{Var}(k_{eff}^{flux}) - \mathrm{Cov}^2(k_{eff}^{coll}, k_{eff}^{flux})\right]. \tag{4.28}$$

The evaluation of the absorption rate follows exactly the same procedure as above and need not be repeated. The same holds for "Expected Leakage Probability" calculations. In this case the estimator $\sum_{j=1}^{J} D_{h,j} W_{h,j-1} + R_{h,J} W_{h,J}$ is replaced by the sum $\sum_{j=1}^{J} (D_{h,j} W_{h,j}^{*} + R_{h,j-1} W_{h,j}^{*LE})$ in equations (4.17) to (4.28).

4.7 Sampling of Time Eigenvalues

The time eigenvalues are calculated at the collision points "j". If $t_{h,j}$ is the time distance at the j'th collision of the h'th neutron history from its origin, the average production time τ_{PR} [Eq. (2.21)] can be calculated according to:

$$\tau_{PR}=\left[\sum_{h=1}^{H}\left(\sum_{j=1}^{J} t_{h,j} W_{h,j-1} P_{PR}\right)\right] \Bigg/ \sum_{h=1}^{H}\left(\sum_{j=1}^{J} W_{h,j-1} P_{PR}\right) \tag{4.29}$$

with the following expression for the variance

$$Var(\tau_{PR}) = \left[\sum_{h=1}^{H}\left(\sum_{j=1}^{J} t_{h,j} W_{h,j-1} P_{PR}\right)^{2}\left(\sum_{j=1}^{J} W_{h,j-1} P_{PR}\right)^{-1}\right] - \tau_{PR}^{2} \tag{4.30}$$

where

$$t_{h,j} = \sum_{g=1}^{j} \frac{D_{h,g}}{\sqrt{E}}.$$

In the same way the average destruction time is given by:

$$\tau_{DE}=\left[\sum_{h=1}^{H}\left(\sum_{j=1}^{J} t_{h,j} W_{h,j-1} P_{AB}\right) + t_{h,J}^{LE} W_{h,J}\right] \Big/ HW_{o} \tag{4.31}$$

where $t_{h,J}^{LE}$ is the time at which the neutron of weight $W_{h,J}$ leaks out of the assembly after the J-th collision:

$$t_{h,J}^{LE} = \left(\sum_{j=1}^{J} \frac{D_{h,j}}{\sqrt{E}}\right) + \frac{R_{h,J}}{\sqrt{E}}$$

and

$$Var(\tau_{DE}) = \sum_{h=1}^{H} \left[\left(\sum_{j=1}^{J} t_{h,j} W_{h,j-1} P_{AB} \right) + t_{h,J}^{LE} W_{h,J} \right]^2 \Big/ HW_o - \tau_{DE}^2 . \quad (4.32)$$

In the case where the "Expected Leakage Estimator" is used, the $W_{h,j-1}$ in Eq. (4.29) has to be replaced by $W_{h,j}^{*}$ and $t_{h,j}$ by

$$t_{h,j}^{*} = \sum_{g=1}^{j} \frac{\Lambda_{h,g}}{\sqrt{E}}$$

The calculation of the mean destruction time τ_{DE} is, however, somewhat different for the E.L.P. method. We have

$$\tau_{DE} = \sum_{h=1}^{H} \left(\sum_{j=1}^{J} t_{h,j}^{*} W_{h,j}^{*} P_{AB} + t'_{h,j} W_{h,j}^{LE} \right) \Big/ HW_o \quad (4.33)$$

where $W_{h,j}^{*}$ is given by Eq. (4.15), $W_{h,j}^{LE}$ by eq. (4.16) and

$$t'_{h,j} = \sum_{g=1}^{j-1} \frac{\Lambda_{h,g}}{\sqrt{E}} + \frac{R_{h,j-1}}{\sqrt{E}} .$$

4.8 The Use of Probability Tables

As already mentioned above TIMOC frequently uses tables with equal probability intervals. It is known that other procedures of calculating random variables are slower than the random selection of precalculated values from equal probability tables. The limits are set by the available storage capacity of the computer. In the input the following random variables are converted into equal probability tables by the Nuclear Data Input (NDI) program: the fission spectra of primary and secondary neutrons, the exponential distribution function of the neutron flight path and the anisotropic elastic scattering distribution. Except for the latter, the selection of the probability intervals is performed by the so-called semi-systematic sampling procedure described in Sec. 4.5. Even though the calculation of equal probability intervals is a trivial problem, a certain sophistication is necessary to obtain a properly functioning computer algorithm.

The G intervals of probability $p_g = \frac{1}{G}$ satisfy the following equations

$$p_g = \frac{1}{G} = \int_{x_{g-1}}^{x_g} f(x)dx \Big/ \int_{x_o}^{x_G} f(x)dx \,, \qquad g = 1,2,\ldots,G \,, \quad (4.34)$$

where $f(x) \geqslant 0$ is the distribution function (either given in analytical form or by table input). The integration is performed by the trapezoid rule. For this purpose the domain $\left[x_o, x_G\right]$ is divided into 200 equidistant intervals

The width of the intervals is:

$$\Delta x' = \frac{x_G - x_o}{200} = x'_g - x'_{g-1} \quad , \qquad g = 1, 2, 3, \ldots, 200 \tag{4.36}$$

and

$$x'_g = x'_o + g\,\Delta x' , \qquad x'_o = x_o , \quad x'_{200} = x_G .$$

To determine the boundaries x_g of the intervals of equal probability summations over g are made until:

$$\frac{\Delta x'}{2} \sum_{g=1}^{k} \left[f(x'_g) + f(x'_{g-1}) \right] - \frac{g}{G} = A_g \geq 0. \tag{4.37}$$

When $A_g \geq 0$ one defines $\tan\alpha_k = \dfrac{f(x'_k) - f(x'_{k-1})}{\Delta x'}$;

then

$$f(x_g) = \sqrt{f^2(x'_k) - 2A_g \tan\alpha_k} \tag{4.38}$$

and

$$x_g = x'_k - \frac{2A_g}{f(x_g) + f(x'_k)} \tag{4.39}$$

The x_g are the boundaries for the intervals of equal probability.

4.9 Random Numbers and Random Variables

The pseudo-random numbers ξ equally distributed in the unit interval (0,1) are generated by the modular method.

To allow reproducibility and to evaluate differential effects in similar problems, a given history always starts with the same pseudo-random number in all problems. This requires a kind of two-dimensional chain of random numbers. In the IBM 7090 the initial element of the i-th chain is determined by

$$2^{35}\xi_o^i \equiv 3^{21}\, 2^{35}\xi_o^{i-1} \pmod{2^{35}} \qquad i = 1,2,\ldots \qquad (4.40)$$

$$2^{35}\xi_o^0 \equiv 1.$$

The successive elements of the chain itself are determined by

$$2^{35}\xi_j^i \equiv 5^{13}\, 2^{35}\xi_{j-1}^i \pmod{2^{35}} \qquad j = 1,2,\ldots \qquad (4.41)$$

In these formulae we have followed the notation of (22) in which the chains were described in terms of integers for convenience. An IBM 7090 number has a sign bit and 35 numeric bits, to be interpreted as a binary fraction: if ξ is such a fraction, $2^{35}\xi$ is an integer. It can be shown that this method has a period 2^{33} which is an upper limit for mod 2^{35}.

5. NEUTRON TRANSPORT ROUTINES

In this chapter we describe the actual routines which calculate the neutron transport and which therefore represent the scattering kernels mentioned above. This description presents the physical models used and the various options offered by the code. Technical details about data input and format are listed in Sec.8.1 and 10.1 but, for those users who may wish to change the code, some mention is made of the entry names of subroutines and the program names of the more important variables.

In general, the computation of histories closely simulates the behaviour of the individual neutrons (although alterations are possible on request through various options to improve the efficiency of the program). Certain isotope and mixture cross sections are required to implement these methods. To calculate actual neutron path lengths, macroscopic mixture cross sections are needed. Hence for each mixture there is a list of the total cross sections as a function of the specified energy group. There is, furthermore, a table which specifies the mixtures in each region. The total cross section for a mixture is the sum of all macroscopic capture, elastic, inelastic, and fission cross sections of all isotopes in the mixture.

Once all the data pertinent to the origin of the history are determined, the neutron travels in the chosen direction until its flight length (or "travel allowance") is exhausted. This is accomplished by using the total cross section in each region encountered. The coordinate location where

the flight is spent is designated as a collision point. At this point various kinds of data are collected and new paramters determined, such as contributions to the calculated flux spectra, absorption and fission processes etc. The kind of scattering is then decided and from that an energy loss and a new direction vector are determined. A new flight length is selected and a similar process is repeated for the next trajectory.

5.1 Initial Parameters of a Neutron History

5.1.1 The Source Neutron Starting Point

The determination of the starting point of primary neutrons is part of the geometry routine. It is therefore described together with the geometry routines. The source region must be specified in the INPUT program. The source distribution is normally flat over the source region. An exception is the use of the STPT * option. In this case the three coordinates of the starting point are input parameters of the RWS program (chap. 10). The specified starting point must lie in the source region specified by the NDP program (chap. 9). The actual position of a particle is referred to in the code as X, Y, Z.

5.1.2 The Initial Direction of Flight

The initial direction of flight of a neutron is assumed to be uniformely distributed over the unit sphere in the lab system. Therefore

$$\cos\phi = \cos 2\pi\xi_g, \tag{5.1a}$$

$$\cos\Theta = 2\xi_{g+1} - 1. \tag{5.1b}$$

The calculation is done in the subroutine ISCLM. An exception is the use of the ANGL * option, where $\cos\Theta$, $\sin\phi$ and the sign of $\cos\phi$ are input parameters. The quantities $\sin\phi$, $\cos\phi$, $\sin\Theta$, $\cos\Theta$ are referred to in the code as SINP, COSP, SINT, COST.

5.1.3 The Length of the Flight Path

In a homogeneous infinite medium the probability density function for the flight length to the next collision point is given by

$$p(x) = \Sigma_{TO} \exp\left(-\Sigma_{TO} x\right), \qquad 0 \leq x < \infty \tag{5.2}$$

where Σ_{TO} is the total macroscopic cross section of the medium. In a piecewise homogeneous finite medium this formula has to be replaced by (see fig. 3)

$$p(x) = \Sigma_{TO}(1) \exp\left[-\Sigma_{TO}(1) x\right] \qquad \text{for } 0 \leq x < x_1, \tag{5.3}$$

$$p(x) = \exp\left\{x_1\left[\Sigma_{TO}(2) - \Sigma_{TO}(1)\right]\right\} \Sigma_{TO}(2) \exp\left[-\Sigma_{TO}(2) x\right]$$

$$\text{for } x_1 \leq x < x_2,$$

$$\vdots$$

$$p(x) = \exp\left\{\sum_{m=1}^{M-1} x_m \left[\Sigma_{TO}(M) - \Sigma_{TO}(m)\right]\right\} \Sigma_{TO}(M) \exp\left[-\Sigma_{TO}(M) x\right]$$

$$\text{for } \sum_{m=1}^{M-1} x_m \leq x < \sum_{m=1}^{M} x_m , \quad M = 3, 4, 5, \ldots .$$

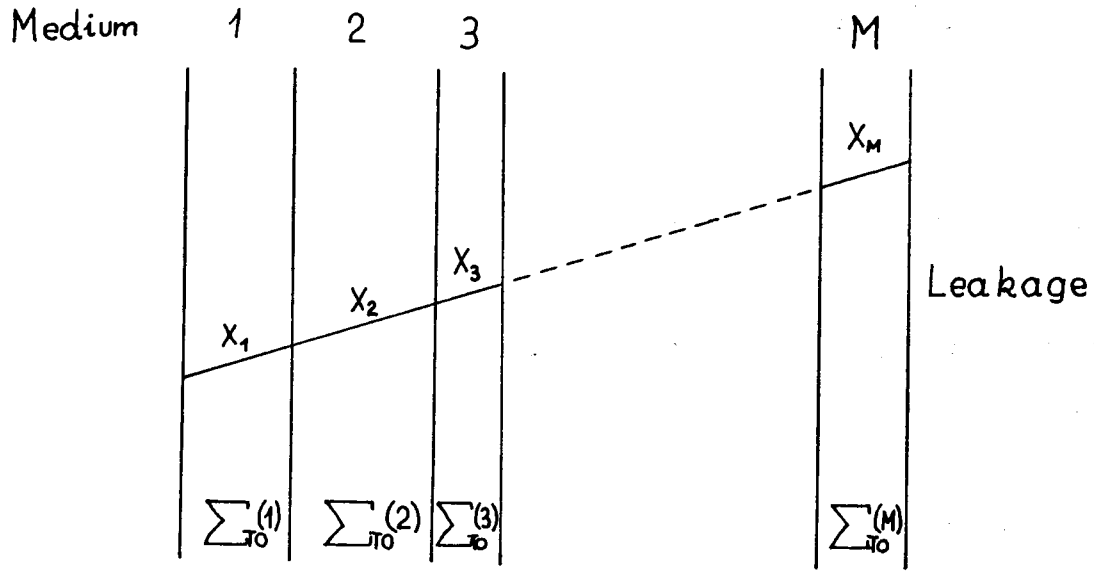


Fig. 3

The function p(x) is discontinuous at the boundaries between two regions if the $\sum_{TO}$ is different. Since it is rather complicated to sample from the function p(x), we transform it into a continuous function over the mean free path using, in each region m, the transformation

$$\sum_{TO}(m)x + \sum_{g=1}^{m-1} x_g \left[\sum_{TO}(g) - \sum_{TO}(m)\right] = \lambda, \qquad (5.4)$$

$$\sum_{TO}(m)\,dx = d\lambda,$$

$$x = \sum_{g=1}^{m-1} x_g, \qquad \lambda = \sum_{g=1}^{m-1} x_g \sum_{TO}(g) = \lambda_{m-1},$$

$$x = \sum_{g=1}^{m} x_g, \qquad \lambda = \sum_{g=1}^{m} x_g \sum_{TO}(g) = \lambda_m.$$

The density function over the mean free path now gets the much simpler form:

$$p(\lambda)d\lambda = \exp(-\lambda)d\lambda, \qquad 0 \leqslant \lambda < \infty. \tag{5.5}$$

The flight distance λ to the next collision point (in mean free paths) is therefore given by

$$\lambda = -\ln \xi. \tag{5.6}$$

If $\lambda \geqslant \lambda_M$, the neutron is lost by leakage.

The collision point is determined from the length of the flight path and the direction of flight. The process occuring as a result of the collision can be either absorption, scattering or fission.

The flight distance λ is determined in the subroutine ISCLM and stored in location MFP. An exception to this is when the ELP * option is used, the next collision point then being always chosen within the reactor so that a neutron history is never terminated by leakage.

The number of mean free paths in the flight distance is chosen from a table of 256 intervals of equal probability.

5.1.4 The Source Energy

The energy distribution of the primary neutrons can either be expressed by one of the three spectra listed below or by a combination of them (see 8.1.1, Block II).

If the STEN* option is used, the source neutrons are supposed to be mono-energetic and the energy has to be specified separately in the input.

The three source energy spectra are:

a) the simple fission spectrum:

$$\chi(E) = B\sqrt{E}\,\exp(-E/A),$$

b) the Maxwell spectrum:

$$\chi(E) = B \cdot E \cdot \exp(-E/A),$$

c) the Watt spectrum

$$\chi(E) = C \cdot \exp(-E/A) \cdot \sinh(\sqrt{B \cdot E}), \qquad (5.7)$$

where E = energy and A, B and C are input parameters of the NDP program. This spectrum is transformed by the NDP program into a historgram with 128 intervals such taht the area of each step is the same. In the actual calculation the intervals are picked by the semisystematic sampling procedure (sec. 4.5). The determination of the starting energy is performed in the subroutine FENP, the energy is stored in the location E.

5.1.5 The Initial Time Parameter

It is assumed that all primary neutrons start at the time point zero. The subroutine INTIM sets this initial time parameter but it can be replaced if required by a routine giving any start-time distribution. The actual time of a neutron history at a reaction point (collision or boundary crossing) is stored in the location TIME and the starting time of a neutron is stored in the location TZERO. The time variable is only calculated, if the option TIME * or the option MTIM * is specified.

5.2 The Transfer to the Geometry Routine (Determination of Boundary Crossing and Collision Point)

When all parameters described in 5.1.1-5.1.5 are determined, the program goes to the subroutine GPATH which is part of the geometry subprogram. The geometry subprogram is an almost independent subroutine which requires only the following parameters to determine collision or boundary crossing points of a neutron trajectory: total reaction cross section = $\sum_{TO}$ (SGT), available travel distance in mean free paths = λ (MFP), Cartesian space coordinates = X, Y, Z, azimuthal angular function $\sin\phi$ and $\cos\phi$ (COSP, SINP) and polar angular function $\sin\Theta$ and $\cos\Theta$ (COST, SINT).

The GPATH routine follows a neutron trajectory through the preassigned geometrical configuration and determines geometrical boundary crossings and collision points as a function of $\sum_{TO}$ and λ. If the travel distance (in mean free paths) calculated by Eq. (5.6) is exhausted along the flight vector before the boundary of the region is reached, a collision process is registered and the coordinates of the collision point are calculated; if the distance to the boundary is shorter than the travel distance, no collision happens in this region. In the latter case, the intersection point with the boundary is determined and the $\sum_{TO}$ of the next region into which the trajectory enters is provided by the RWS program. During this brief return to the RWS program, calculations which make use of the travel distance to the boundary are also performed as will be described later. If the new region into which the neutron enters is the leakage region, the current history is terminated and the weight of the neutron is added to the total leakage register.

When a collision takes place, the type of collision is determined and the new direction of the flight vector and energy has to be calculated. A new travel distance has also to be assigned to the history under consideration. If the energy group was also changed, a new $\sum_{TO}$ has to be stored in SGT before the program returns again to the GPATH routine.

5.3 Reactions at a Collision Point

At a collision point, the program decides which kind of scattering takes place. For this purpose, the "NDP Prgram" prepares a list of elastic and inelastic scattering probabilities for each isotope k as a function of the energy groups i. By adding up these probabilities and comparing them with a random number, ξ , in the interval (0,1), the program determines the kind of collision (elastic or inelastic) and the isotope involved in the process. The probabilities are defined as:

$$P_{EL}(k,m,i) = \sum_{EL}(k,m,i) \Big/ \sum_{SC}(m,i) \tag{5.8}$$

and

$$P_{IN}(k,m,i) = \sum_{IN}(k,m,i) \Big/ \sum_{SC}(k,m,i) \tag{5.9}$$

where $\sum_{EL}$ and $\sum_{IN}$ are the macroscopic elastic and inelastic scattering cross sections and $\sum_{SC}$ the total scattering cross section of the mixture m in the energy group i.

5.3.1 Elastic Scattering (Isotropic and Anisotropic)

If elastic scattering occurs, four types of scattering model can be used optionally for each isotope and energy group. a) Isotropic scattering in the center of mass system; b) Anisotropic scattering in the center of mass system inserted by means of the first six coefficients of a series of Legendre polynomials; c) Anisotropic scattering in the center of mass system inserted by means of an equally spaced table of numbers giving the μ dependence of the differential cross section $\frac{d\sigma_{EL}}{d\mu}$ (the code automatically renormalizes these numbers so that the integral over μ fits the independently given total scattering cross section); d) Anisotropic scattering in the laboratory system inserted by means of the average cosine of the scattering angle in the laboratory system ($\bar{\mu} = \overline{\cos\vartheta}$).

In the following discussion it is always assumed that the target nucleus is at rest. This is valid for all neutron energies, except in the thermal region, where the thermal motion of the target can be taken into account by means of transfer matrices which include the energy transfer from the target nucleus to the neutron.

The above mentioned scattering models allow the evaluation of the angular deflection of the velocity vector of the neutron due to scattering and deal always with the angle ϑ measured in the scattering plane.

In the case of isotropic scattering in the center of mass system the values of the function $\cos\vartheta$ are equally distributed in the interval (-1, +1), i.e.

$$\cos\vartheta = 2\xi - 1.$$

This and the associated calculation of the energy loss and the transformation to the laboratory system are performed in the ECO-routine.

In the case of anisotropic scattering b) and c) the $\cos\vartheta$ values are picked at random from a probability table. Both the Legendre Polynomial and the tabulated distribution are transformed into a histogram in which each step has the same area (probability). Depending on the input specifications, this histogram is represented as a table containing 16 or 32 equal probability intervals, i.e.:

$$\frac{1}{16}\left(\text{or } \frac{1}{32}\right) = \int_{\cos\vartheta_g}^{\cos\vartheta_{g+1}} d(\cos\vartheta)\, f(\cos\vartheta) \Big/ \int_{-1}^{1} d(\cos\vartheta)\, f(\cos\vartheta) \tag{5.10}$$

where $\cos\vartheta$ is calculated from

$$\cos\vartheta = \cos\vartheta_g + \xi_2(\cos\vartheta_{g+1} - \cos\vartheta_g) \tag{5.11}$$

and

$$g = \left[16\,(\text{or } 32)\,\xi_1\right].$$

The formalism for setting up the above tables is described in sec. 4.8.

In addition to the angle of deflection, ϑ, from the original line of flight, a second angle φ has to be calculated which has a uniform distribution function around the incident line of flight. This is given by:

$$\cos\varphi = \cos 2\pi\xi \tag{5.12}$$

or, if a rejection technique is used, by

$$\cos\varphi = \frac{(2\xi_1-1)^2 - \xi_2^2}{(2\xi_1-1)^2 + \xi_2^2} \tag{5.13}$$

where, unless the condition

$$(2\xi_1-1)^2 + \xi_2^2 \leq 1 \tag{5.14}$$

is satisfied, (5.13) is recalculated with two new random numbers ξ_3 and ξ_4.

The energy E' of the neutron after an elastic scattering is a function of $\cos\vartheta$ and the atomic weight A(k). If the collision process is treated in the center of mass system the following well known relation holds:

$$E' = E\,\frac{A^2(k) + 2A(k)\cos\vartheta + 1}{(A(k)+1)^2}\,. \tag{5.15}$$

In the case of an isotropic scattering on a hydrogen atom this reduces to the simple form

$$E' = E\,\xi \tag{5.16}$$

which is separately programmed in the ECO-routine.

If E' is less than the lower energy limit specified in the problem, the history is finished. If not, the angular components of the new flight vector after the collision have to be determined.

For this purpose one has first to transform the angle of deflection from the c.m. system into an "intermediate" lab. system with the z-axis in the incident line of flight:

$$\cos\Theta = (1+A(k)\cos\vartheta)/\sqrt{1+2A(k)\cos\vartheta+A^2(k)}. \tag{5.17}$$

The isotropic distribution of φ around the incident line of flight remains in this lab. system the same as in the c.m. system. Denoting the polar coordinates of the initial flight path in the real laboratory system by (ϑ_1, φ_1), it follows that the transformation into the real lab. system can be accomplished by a rotation about the z-axis by the angle φ_1 followed by a rotation about the y-axis by the angle ϑ_1. The new direction vector in the real lab. system is thus given by the unit vector:

$$\begin{pmatrix} e_1 \\ e_2 \\ e_3 \end{pmatrix} = \begin{pmatrix} \cos\varphi_1 & -\sin\varphi_1 & 0 \\ \sin\varphi_1 & \cos\varphi_1 & 0 \\ 0 & 0 & 1 \end{pmatrix} \begin{pmatrix} \cos\vartheta_1 & 0 & \sin\vartheta_1 \\ 0 & 1 & 0 \\ -\sin\vartheta_1 & 0 & \cos\vartheta_1 \end{pmatrix} \begin{pmatrix} \overline{e_1} \\ \overline{e_2} \\ \overline{e_3} \end{pmatrix}. \tag{5.18}$$

The direction coordinas $\begin{pmatrix} \overline{e_1} \\ \overline{e_2} \\ \overline{e_3} \end{pmatrix} = \begin{matrix} \sin\Theta & \cos\phi \\ \sin\Theta & \sin\phi \\ \cos\Theta & \end{matrix}$ refer to the "intermediate" lab. system.

From the components of the new direction vector $\vec{e}$ with the angular components ϑ_2 and φ_2 one can calculate the trigonometric angular functions as:

$$\begin{aligned}
\cos\vartheta_2 &= \cos\vartheta_1\cos\Theta - \sin\vartheta_1\sin\Theta\cos\phi, \\
\sin\vartheta_2 &= \sqrt{1-\cos^2\vartheta_2}, \\
\sin\vartheta_2\cos\varphi_2 &= \cos\varphi_1(\cos\vartheta_1\sin\Theta\cos\phi+\sin\vartheta_1\cos\Theta)-\sin\varphi_1\sin\Theta\sin\phi, \\
\sin\vartheta_2\sin\varphi_2 &= \sin\varphi_1(\cos\vartheta_1\sin\Theta\cos\phi+\sin\vartheta_1\cos\Theta)+\cos\varphi_1\sin\Theta\sin\phi.
\end{aligned} \tag{5.20}$$

When the scattering in the c.m. system is isotropic (option (a) above) and the atomic weight A> 50 the scattering in the lab. system is also assumed to be isotropic and the above transformation is bypassed. In this case, the scattering angles Θ and ϕ are determined by the ISCLM routine.

If anisotropic scattering is inserted by means of the average cosine $\bar{\mu}$ (option (d) above) a procedure is used which is based on a technique developed by Coveyou. This translates $\bar{\mu}$ back into a linear anisotropy function in the lab. system from which discrete cos Θ values are sampled. The energy loss in this case expressed is given by

$$E' = E\,\frac{\left[\cos\Theta+\sqrt{A^2(k)-1+\cos^2\Theta}\right]^2}{\left[A(k)+1\right]^2}. \tag{5.21}$$

5.3.2 Scattering with Energy-Angle Change Specified by a Transfer Matrix

For a transfer matrix description, a list of scattering probabilities to the specified energy groups is prepared by the NDP program. By using equally distributed random numbers, the energy intervals to which scattering occurs are chosen according to their probability values.

Since the program works with discrete energies (if no other option is specified) the energy is chosen to be equally distributed within the group. When the transfer matrix is used, neutron production processes like (n,2n) reactions, etc. can also be taken into account. If the sum of the differential cross sections of the scattering matrix (in a certain energy group) is not equal to the total inelastic scattering cross section of the specified isotope in this group,a multiplication factor is defined which is applied to the neutron weight. This multiplication factor is calculated as the ratio

$$r(k,i) = \sum_{g=1}^{G} \Delta\sigma_{IN}(k,i\to g) \Big/ \sigma_{IN}(k,i) \qquad (5.22)$$

where $\Delta\sigma_{IN}(k,i\to g)$ is the inelastic scattering cross section of isotope k for IN neutron transfer from energy group i to energy group g. For a multiplication factor r(k,i) different from unity, the neutron weight $W_{h,j}$ is multiplied by r(k,i) after the occurrence of an inelastic collision process. This fact has to be considered when checking the neutron balance at the end of a calculation because, with $r(k,i) \neq 1$ the sum of neutron losses (by leakage, absorption and eventually slowing down below a preassigned energy level) does not correspond to the number of neutrons originally introduced into the system.

In order to take into account anisotropy, which is sometimes of interest for inelastic scattering processes, one can assign to each partial transfer cross section an averaged $\overline{\cos\Theta}$ value, defined for the lab. system. This average $\overline{\cos\Theta}$ value is transformed into a linear anisotropy from which discrete $\overline{\cos\Theta}$ values are sampled. This and the transformation of the direction vector at the collision point is performed by the TURN routine.

5.3.3 Inelastic Scattering by Excited Levels

When the model of excited energy levels is employed, use is made of a table containing the partial probabilities $p_g^{lev}(k,i)$ of the different levels in a certain energy group. The table is prepared by the NDP program for the different isotopes. The energy loss ΔE_g^{lev} associated with each level is subtracted from the energy of the incident neutron after the level has been chosen by a random process.

If $\xi - p_1^{lev}(k,i) < 0$ then the first excited level is used and

$$E' = E - \Delta E_1^{lev}; \tag{5.23}$$

if not, try:

$$\xi - p_1^{lev}(k,i) - p_2^{lev}(k,i) < 0 \tag{5.24}$$

and so on.

$$\sum_{g=1}^{G} p_g^{lev}(k,i) = 1 \tag{5.25}$$

where G is the number of excited levels.

The use of discrete neutron energies within an energy group in combination with the model of level excitation may result in a negative energy value after an inelastic scattering process, i.e. $E' = E - \Delta E_{g\,max}^{lev} < 0$. If this occurs, another level is chosen. If however there is no level which satisfies the condition $E' - \Delta E_{g}^{lev} \geqslant 0$, then the program performs an elastic scattering process using the same isotope.

For the model of level excitation the angular distribution of the scattered neutrons is assumed to be isotropic in the lab. system.

5.3.4 Inelastic Scattering by the Evapporation Model

The Statistical or Evaporation Model assumes a Maxwellian energy distribution function for the scattered neutrons. The energy after the collisions is taken from the following probability density function:

$$N(E')dE' = \frac{E'}{T^2(k)} \exp\left[-E'/T(k)\right]dE' \qquad (5.26)$$

where E' is the neutron energy after the collision and $T(k) = c(k,i)\sqrt{E}$ the so-called nuclear temperature composed of the constant c(k,i) and the incident neutron energy E. This model is frequently used for heavy isotopes at higher energies where the levels cannot be resolved.

A Maxwellian distribution can easily be generated by the sum of two exponential distributions and this is done in the subroutine ICMXW. The energy of the scattered neutron is then

$$E' = T(k)(\ln\xi_1 + \ln\xi_2). \qquad (5.27)$$

If $E' > E$ the random process has to be repeated.

5.3.5 Absorption

The absorption as well as the fission process is taken into account by the "method of weights" or the so-called "method of expected values". In this model one assigns to each neutron history at its beginning a weight factor W_o. This weight W_o is usually set to unity. At each collision point the weight factor is multiplied by the probability of its survival P_{sc}:

$$W_{h,j} = P_{sc} W_{h,j-1} = W_o \prod_{g=1}^{j} P_{sc,g}(m,i) \tag{5.28}$$

where j is the collision counter in the h-th history and

$$P_{sc}(m,i) = \sum_{k=1}^{K} \left[\sum_{EL}(k,m,i) + \sum_{IN}(k,m,i) \right] \Big/ \sum_{TO}(m,i) \tag{5.29}$$

is the scattering probability of mixture m in the energy group i.(The absorbed weight $W_j^{AB} = W_{j-1} - W_j$ is accumulated for each spatial region l as a function of the energy group i and is part of the output.)

5.3.6 Fission

The TIMOC code treats fission processes by the so-called "method of fractional generated neutrons". This procedure is explained above in Sec. 4.3 ("Variance Reducing Methods") and the operation of the two basic options of the program are described in chapter 6 ("Calculation Schemes") which also contains schematic flow diagrams. The calculation of the fractional neutrons $\psi_{h,i}(k)$ at each collision point is performed in the subroutine FISWS.

5.4 The Termination of a Neutron History

In an analogue Monte Carlo game a neutron history is terminated either if the neutron leaves the phase space over which the integration is performed (leakage, slowing down etc.) or if it is absorbed. The introduction of variance reducing methods like the method of expected values (Standard Weight Estimator or the Method of Expected Leakage) changes this direct picture of nature and requires the introduction of other criteria for the termination of neutron histories. TIMOC provides the following possibilities for terminating histories. They can be used in any meaningful combination.

5.4.1 Termination by Leakage

In finite geometries and if ELP* is not specified, a history is terminated when the neutron track enters a leakage region (defined in the input). At this point several sample values are calculated and stored, such as the leakage weight and the time at which leakage occurs after the neutron birth. For more information see 10.4.3, 10.4.9, If ELP* is specified and TIMOC works in the Expected Leakage Version the histories are terminated either by the Russian Roulette procedure or, in time dependent problems, when the time from birth exceeds a pre-assigned parameter t_{max} (see Sec. 5.4.4).

5.4.2 Termination by Russian Roulette

If ELP* is not specified, the termination by Russian Roulette is in many cases a variance reducing modification of the termination by absorption (Sec. 4.2).

5.4.3 Termination by Slowing Down

In slowing down calculations where only a certain energy range is of interest, or in Fermi Age calculations, a neutron history is terminated after a collision process if the energy assumes a value which is below an E_{min} specified in the input. The slowing down density and other pertinent sample values are printed in the output (see Sec. 10.4.4, 10.4.5, 10.4.9).

5.4.4 Termination by Exceeding a Maximum Time Parameter

If TMAX* is specified, neutron histories are terminated if their time from birth exceeds at a boundary crossing or at a collision point the value t_{max}. See 10.1 and 10.2 card 8. This feature is particularly useful if at the same time TIME* and ELP* are specified (calculation of time dependent neutron migration in small systems). See 4.4 and (10).

5.4.5 Termination by Low Weight

Finally a neutron history is terminated if its weight becomes too small and does not contribute significantly to the result anymore. This cut off value depends on the problem under consideration. The cut off boundary built into the code is 10^{-3} and is stored in 1FLM3. If desired, a change of this quantity can be accomplished easily by the use of the COMM* specification. See 10.1 and 10.2 card 11.

6. THE TREATMENT OF SECONDARY NEUTRONS AND CALCULATIONAL SCHEMES

It should be noted that all neutrons which do not belong to the primary neutron source or source distribution fall under the term "secondaries" regardless of their actual generation number. As mentioned before the fractional generated neutrons are stored for each fissionable isotope separately. These "secondaries" have a spatial distribution function which depends on the fission probability of their ancestors.

The initial direction vector of the secondary neutrons is chosen to be isotropic in the laboratory system. The initial energy of secondary neutrons is taken from the specific energy distribution function of the isotope under consideration. The energy distribution function are the same as for the source neutrons (see 5.1.4), i.e. the simple fission spectrum, the Maxwell spectrum, the Watt spectrum or a combination of them. The specific parameters are part of the nuclear data input in NDP (see 8.1.1, Block II).

In order to generate this energy distribution, tables of 128 equal probability intervals are prepared for each fissionable isotope and the intervals are then chosen by semisystematic sampling. As usual, a linear random interpolation within each interval is performed.

To make correct use of the TIMOC code one has clearly to distinguish between thos calculations in which a constant external source (in the case of a stationary problem) or a δ-burst (in the case of a time dependent study) is of interest, and those in which the equilibrium or fundamental mode distribution is important. The two schemes differ in the treatment

of the secondary neutrons generated in the FISWS-routine. We describe them here by the use of schematic flow diagrams (for a system with one fissionable isotope in order to omit the index k).

6.1 Initial Value Problems (Standard Version)

In the case of a constant external source in stationary problems or a δ -burst in the case of a time dependent problem, the scheme of Fig. 4 for generating and sampling secondary neutrons is used (this is the basic version of the code). In this version, primary neutrons start from a pre-assigned energy, angle and space distribution. Whenever a secondary neutron of unit weight is generated (i.e. $\sum \psi_{h,j} \geq 2\xi W_0$, see Fig. 4), its position and other parameters are stored in a buffer (B in Fig. 4), while the current history is taken to its completion. However, before a new neutron is inserted from the source, the neutron or neutrons accumulated in the buffer (B) are introduced into the calculation as additional histories. Amost all sample values are normalized by the number of primary histories and therefore the sum of destroyed neutrons (by leakage, absorption and eventually slowing down below a specified energy limit) adds up to unity only in the case where non fissionable materials are used.

6.2 Eigenvalue Calculations (The "ITER" Version)

In the case of an eigenvalue calculation, i.e. in the case where the sample values should be obtained from the equilibrium or so-called fundamental mode distribution, the iteration (ITER) version of the code has to be em-

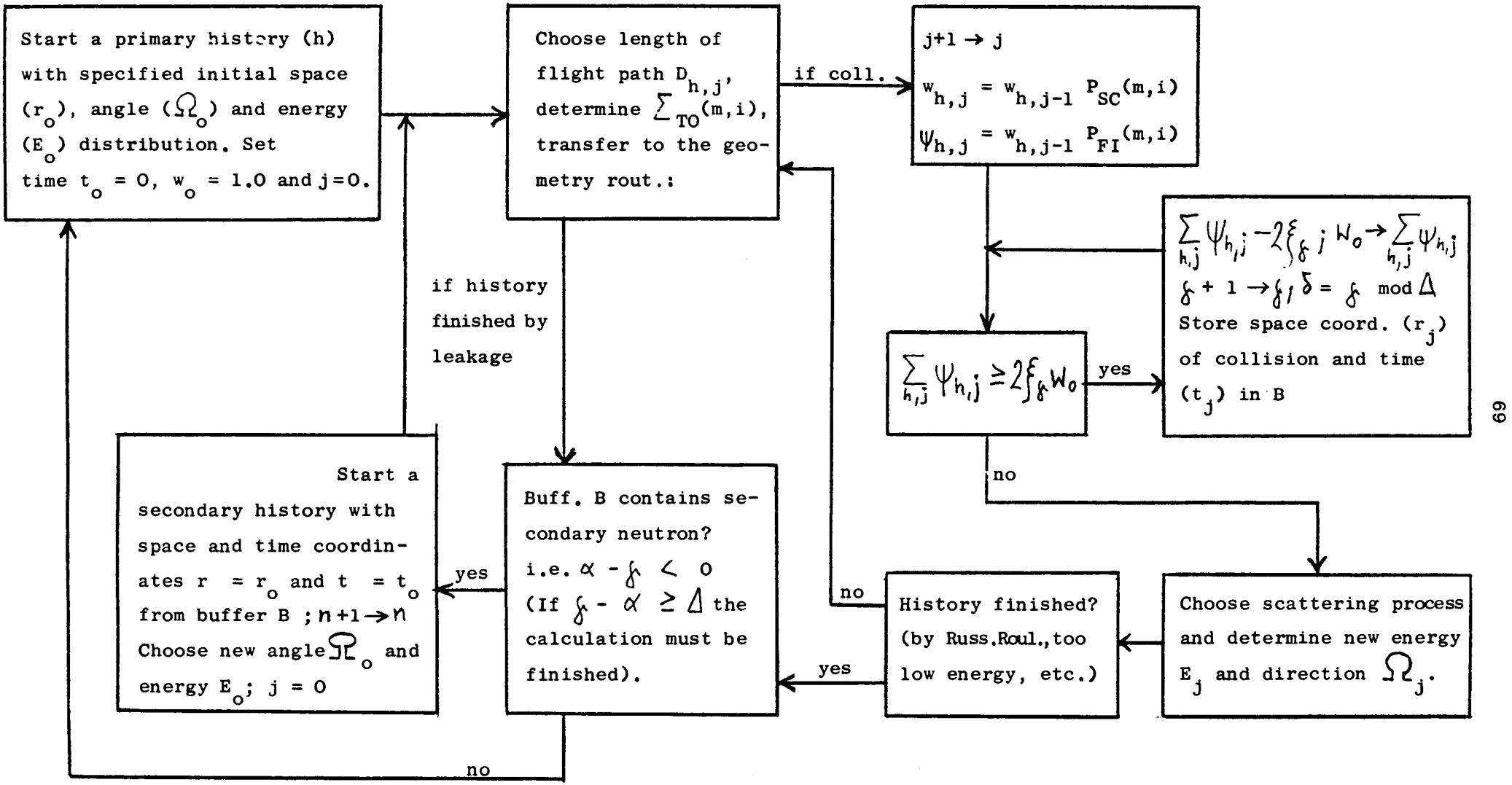


Fig. 4 Schematic flow diagram for the solution of initial value problems ("Standard" Version) in TIMOC

ployed. In this version one first starts a certain number of histories in the basic version of the codes as described above. The source distribution of these neutrons should be not too different from the final equilibrium distribution (for example: fission neutrons spatially equal distributed over the fission zone). These neutron histories are not used for sampling. They only build up a second neutron generation which is closer to the final equilibrium distribution and which is used to start the sampling procedure. These neutrons stored in the buffer (B in Fig. 5) are "removed" as the calculation proceeds from the buffer store (B_β) and replaced by newly generated ones in the free buffer locations under or overflow of the buffer store (i.e. $\delta-\beta<0$ or $>\Delta$ where Δ is the buffer length) irrespective of whether the system under consideration is subcritical or supercritical. To avoid this, one uses the multiplication factor as a test parameter to decide when the characteristic parameters of a secondary neutron have to be preserved in the buffer storage. Since the multiplication factor is not known at the beginning of the calculation, the program uses an estimate provided by the histories already calculated. This estimate is recalculated later, whenever the buffer storage shows a tendency for under or overflow. In the case of underflow, which means that too few buffer storages are filled with initial parameters for neutrons or in other words the buffer index γ is approaching α, the test parameter which is obviously too large has to be recalculated. For this purpose one determines k_{eff}^{coll} and subtracts the variance in order to be on the safe side. If there is the tendency for overflow and the difference between γ and α increases towards the length of the buffer storage, the same recalculation of the test parameter has to be performed, but in this case one adds the variance to the multiplication

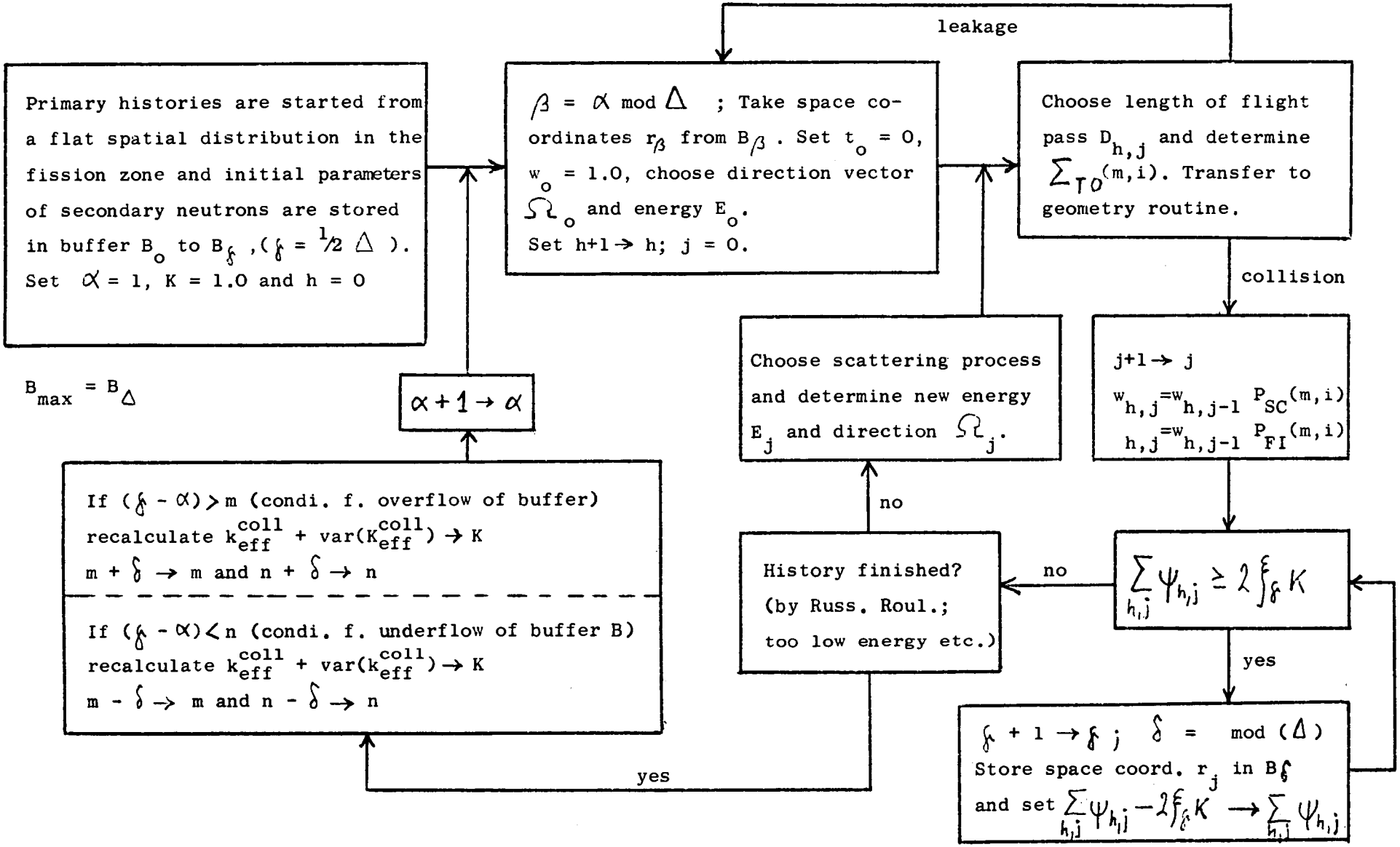


Fig. 5 Schematic flow diagram for Eigenvalue calculations ("ITER" Version) in TIMOC

factor k_{eff}^{coll} . By this procedure one can achieve any number of iterations in the previously described sense of the "source iteration procedure" even for strongly sub or supercritical systems.

6.3. Small Effect Calculations (The "SMEC" Specification)

The TIMOC code is also equipped with a special feature allowing the estimation of small perturbation effects caused for example by the exchange of a control rod or a fuel element. In such calculations the variance of the differential effect does not depend on the total variance of the characteristic parameters like the reactivity, lifetime etc. This special feature therefore makes it possible to independently determine differential effects for all results provided by the code.

These Small Effect Calculations (SMEC * option) are achieved by the known method of similar flight paths. Each neutron history begins with a new chain of random numbers which is the same for both the perturbed and unperturbed case. In the presence of fissionable materials one "marks", so to say, neutron histories after they touch the region of disturbance, in order to get the same spatial distribution of the secondary fission neutrons in both the perturbed and the unperturbed cases. Secondary neutrons produced by the part of a neutron history which has been "marked" are accumulated separately so that they do not disturb the original fission distribution. After a neutron history has been "marked" it is no longer subject to the Russian Roulette procedure, irrespective of whether the RURU option is used or not. A "marked" neutron history is finished only by leakage, a too low weight factor or an energy value which is below the preassigned threshold. See also 10.4.7.

7. THE TECHNICAL STRUCTURE OF THE PROGRAM AND OPERATION PROCEDURES

The program is split up into three parts which are executed consecutively in order to allow for a maximum of data space in the computer memory. The data communication between these program parts is done via an "Intermediate Tape". The execution sequence is (1) Nuclear Data Preparation, (2) Random Walk Sampling, and (3) Time Tape Analysis.

Such a system is usually called a link chain program.For TIMOC it is designed to run with an IBM 7090 Fortran II monitor system using a program "System Tape" which contains the basic microscopic cross sections including nuclear parameters in addition to the program links.

Each file written in binary mode contains either one of the link programs (Data Preparation, Random Walk Sampling or Time Tape Analysis Program) or a cross section library. Thus it is possible to have many cross section libraries and variations of the link programs on the same physical tape. During execution the desired library or link program is called in by the proper file number.

a) The Nuclear Data Preparation Program (link 1) is written mainly in Fortran II. It searches the designated library for the required library data, generates macroscopic cross sections, angular distributions, transfer matrices or level excitations, probability distribution tables which depend on the specified isotropic data, etc. In general it does all preliminary calculations which are geometry independent and writes the results on the "Intermediate Tape" (B5).

b) The Random Walk Sampling Program (link 2) is the actual Monte Carlo part in combination with the geometry routine written in FAP language in order to make full use of the logic of the computer and to speed up the calculations. It processes geometry input data and stores the data prepared by the "Input Program". It follows the neutron histories, performs the sampling procedure and records the results. Periodically it puts a rerun dump on the "Intermediate Tape". If more time parameters than the time eigenvalues (mean generation time and mean destruction time) are required, i.e. if a full time dependent problem is studied, a "Time Tape" is also prepared during the execution of the "Random Walk Sampling Program". It contains information on the time dependent neutron transport parameters and is processed later on by the "Time Tape Analysing Program" (TIME TAPE on B6).

C) The "Time Tape Scanning Program" is a FAP program. It can be used to analyse the neutron histories with respect to the time variable or, in other words, the time dependent die-away of a neutron population in an assembly as a function of position and energy.

In addition to these three programs there exist two Tape Writing Routines which are needed to prepare the "System Tape".

One is the "Library Tape Writing Routine" written in Fortran II which reads the microscopic cross sections and the basic nuclear parameters from the input tape or an updating tape and writes them, after numerous tests for consistency and order, in binary form on the specified file of the "System Tape".

The other is the Program Tape Writing Routine which writes the Data Preparation, Random Walk Sampling and Time Tape Scanning Program on the specified file of the "System Tape".

The above mentioned Programs consist of the following non monitor routines and subroutines. In most cases the binary decks must be put together in the sequence listed below in order to ensure a proper functioning of the program.

Name of link program	Name of Routines	Label of Bin.deck	Label of Fortran or FAP deck	Input
Library Tape Writing	LTW	LTW	LTW	Input 8.1
Routine; LTW	FAP s.r.s. of LTW	LTWFAP	LTFP	(file number and nuclear data)
Nucl. Data Preparation	SPACE	SPACE	SPACE	Input 8.2.1
Program; NDP	BEGIN	BEGIN	BEGIN	(file number)
	START	NDPST	NDST	
	NDPMA	NDPMA	NDMA	
	NDPFI	NDPFI	NDFI	
	EQPT	NDPEQ	NDEQ	
	NDPEL	NDPEL	NDEL	
	PL	NDPPL	NDPL	
	NDPIN	NDPIN	NDIN	
	ERROR	NDPER	NDER	
	FAP s.r.s. of NDP	NDPFAP	NDFP	
	PTW	PTW	PTW	
Random Walk	SPACE	SPACE	SPACE	Input 8.2.1
Sampling Program	BEGIN	BEGIN	BEGIN	(file number)
RWS	WPKO6-7 [3]	WPK67	WPK	
	RWS	RWS	RWS	
	Geom.Rout. [1]	..... [2]	..... [2]	
	PTW	PTW	PTW	
Time Tape Scanning	SPACE	SPACE	SPACE	
Program TTS	BEGIN	BEGIN	BEGIN	
	WPKO6-7 [3]	WPK67	WPK	Input 8.2.1
	TTS	TTS	TTS	
	PTW	PTW	PTW	
LOADER	LOADE	LOADER	LOAD	Input 9.1 and, or 10.1 and, or 11.1 or 10.7

1) At this place the geometry routines are inserted

2) For description see (25)

3) Ref. (27)

8. THE TAPE WRITING ROUTINES PREPARING THE SYSTEM TAPE

The data input consists of two steps. One is the preparation of the System Tape (T = A5) by the use of the Tape Writing Programs and the second is the actual calculation of the sample problem described in Chapter 9. The four subsequently described writing procedures of the System Tape can be executed in any order.

8.1 The Library Tape Writing Routine

Writes the group constants on the System Tape (A5)

8.1.1 Input

CARD	COLUMNS	FORMAT	SYMBOL
1	1 - 6	I6	INTAP
	7 -12	I6	INFIL
	13-18	I6	NFILSY
	19-24	A6	ADDATA
	25-30	A6	UNLDK5
	31-36	A6	UNLDK9

INTAP: Tape number of the nuclear parameter input.

If INTAP = 5: the nuclear parameters are read from the Monitor input tape, i.e. they follow on cards behind card no. 1

If INTAP = 11: the nuclear parameters are read from a special input tape, which is on tape unit no. 11 (A6). In this case the nuclear parameters have to be written on this special input tape by an updating procedure.

CARD	COLUMNS	FORMAT	SYMBOL

INFIL: If INTAP $\neq$ 5: INFIL specifies the file number on tape 11 (A6) containing the nuclear parameters which follow, otherwise it is ignored.

NFILSY: File number where the cross sections should be placed on tape 9 (A5).

ADDATA: if blank: normal loading

if not blank: nuclear parameters for new isotope specifications are added to an already existing library on tape 9 (= A5).

UNLDK5: if blank: tape 11 (A6) is rewound

if not blank: tape 11 (A6) is unloaded

If ADDATA is blank:

2	1 - 6	I6	IM
IM: Number of energy groups, $1 \leq IM \leq 50$.			
3	1 - 11	E11.4	ENE(I) = 1, IM + 1. One card for each number.

ENE(I): The lower energy limits (in eV) of the IM energy groups in increasing order. ENE(IM+1) = upper limit of the top group. Note that the boundaries have to be the same for all isotopes. The above set of cards is required once. The following cards 4.) to 12.) have to be repeated for each isotope.

CARD	COLUMNS	FORMAT	SYMBOL	

If ADDATA is not blank the energy groups have to be omitted, since they are not needed for adding data.

All the following nuclear data and group averaged cross sections can be obtained from the ENDF/B data file in the required Formats by the use of the CODAC code (Ref. 26).

Block I: Parameters which are independent of the energy-group structure

4	1 - 6	A6	ISOT	Isotope identification
5	3(1-72)	3(12A6)	TEXT	

TEXT: three cards, which contain a description of this isotope (origin of the cross sections, etc.; the cards may also be left blank).

6	1 - 6	A6	ISOT
	7 -17	E11.4	ATW
	18-23	I6	IMF
	24-34	E11.4	FNY
	35-45	E11.4	AFIS
	46-56	E11.4	BFIS
	57-67	E11.4	CFIS

CARD	COLUMNS	FORMAT	SYMBOL

ISOT: Isotope identification

ATW: Atomic weight of the isotope

IMF: Number of different fission spectrum representations to be used (≤ 3)

The energy dependence of the number ν of secondary fission neutrons is assumed to be described by the following polynomial fitting:

$$\nu(E) = FNY + AFIS*E + BFIS*E^2 + CFIS*E^3$$

E = energy (eV).

$\nu(E)$ can also be given as a group averaged value for each energy group separately" see card 8.

7	1 - 6	I6	LTT
	7 -17	E11.4	EMIN
	18-28	E11.4	EMAX
	29-39	E11.4	ELCO(1)
	40-50	E11.4	ELCO(2)
	51-61	E11.4	ELCO(3)

LTT: Symbol defining type of fission spectrum

EMIN, EMAX: Lower and upper limit (in eV) for the corresponding fission spectrum

ELCO(1): A

ELCO(2): B

ELCO(3): C

LTT = 6 or 7: Simple fission spectrum

$$\chi(E) = B\sqrt{E}\exp(-E/A)$$

CARD	COLUMNS	FORMAT	SYMBOL

LTT = 8 or 9: Maxwellian distribution

$$\chi(E) = B \cdot E \cdot \exp(-E/A)$$

LTT = 10: Watt spectrum

$$\chi(E) = C \cdot \exp(-E/A) \cdot \sinh(\sqrt{B \cdot E})$$

There can be a maximum of 3 cards of type 7 per isotope.

Block II: All microscopic group averaged cross sections split into the capture, elastic scattering, inelastic scattering and fission parts and (optional) the particle multiplication factor for fission.

8	1 - 6	A6	ISOT
	7 -17	E11.4	ENCH
	18-28	E11.4	CROM(1)
	29-39	E11.4	CROM(2)
	40-50	E11.4	CROM(3)
	51-61	E11.4	CROM(4)
	62-72	E11.4	CROM(5)

ISOT: Isotope identification

ENCH: Lower boundary of the energy group (eV)

CARD	COLUMNS	FORMAT	SYMBOL

CROM(1): is σ_{CA}, the microscopic capture cross section
Unit: barn (= 10^{-24} cm^2).

CROM(2): is σ_{EL}, the microscopic elastic scattering cross section
Unit: barn.

CROM(3): is σ_{IN}, the microscopic inelastic scattering cross section
Unit: barn.

CROM(4): is σ_{FI}, the microscopic fission cross section
Unit: barn.

CROM(5): is ν (like on card 6). If CROM(5) $\neq$ 0, this value is taken for the determination of the product $\nu \cdot \sigma_{FI}$, but even in this case a card 6 is obligatory in the library deck. In any case the parameters FNY,..., CFIS of card 6 are used to calculate the fission ratio (Fissions per Primary Neutron) in the final print-out of a calculation.

Block III: All information on elastic isotropic or anisotropic scattering

A card 9 must be present for each energy group (in increasing order) in which $\sigma_{EL} \neq 0$. If required, card 9 must be followed by the corresponding card 10. The card(s) describe(s) the differential cross sections for the elastic anisotropic scattering.

9	1 - 6	A6	ISOT
	7 -17	E11.4	ENCH
	18-23	I6	LTT
	24-29	I6	NE

ISOT: Isotope identification

ENCH: Lower boundary of the energy group (eV)

LTT: Symbol defining angular distribution description

NE: Number of parameters following on card 10

CARD	COLUMNS	FORMAT	SYMBOL

LTT = -1: The same angular distribution function as in the previous group is used. Card 10 must be omitted.

LTT = 0: Isotropic scattering distribution in the c.m. system. Card 10 must be omitted.

LTT = 1: The distribution is described by a Legendre polynomial in the c.m. system. The program expects on card 10 NE Legendre coefficients for the anisotropic distribution

$$\frac{d\sigma}{d(\cos\vartheta)} = \sum_{n=0}^{NE-1} ELCO(n+1)\cdot P_n(\cos\vartheta).$$

LTT = 2: Anisotropic scattering is described in the c.m. system by a polygon. NE is the number of equidistand points along the μ-axis in the $d\sigma/d\mu$ table (≤ 40).

LTT = 3: The averaged value cos θ of the angular distribution in the lab. system is used.

Card(s) 10 must only be given, if LTT $\geq$ 1.

10	1 -11	E11.4	ELCO(1)	
	.	.	.	
	.	.	.	ELCO(I),I = 1,NE
	. .	.	.	
	56-66	E11.4	ELCO(6)	
	etc.			

IF LTT = 1: The ELCO(I) are the NE coefficients of the Legendre polynomial.

CARD	COLUMNS	FORMAT	SYMBOL

If LTT = 2: The ELCO (I) are the NE values (≤ 40) of the angular distribution $d\sigma/d\mu$ at equidistant points between $[+1,-1]$, the boundaries included. The number of intervals is therefore NE-1. The ELCO(I) must be given for the distribution in the c.m. system and in decreasing order of $\cos\vartheta\left[+1 \rightarrow -1\right]$.

Note that in each group another type of anisotropic scatterming may be used.

Block IV: All information on inelastic scattering and transfer matrices

11	1 - 6	A6	ISOT
	7 -17	E11.4	ENCH
	18-23	I6	LTT
	24-29	I6	NU

ISOT: Isotope identification

LTT defines the inelastic scattering description

ENCH: Lower energy boundary

NU: Number of parameters following on card 12

LTT = -1: The same scattering values are used as in the previous energy group.

LTT = 1: The model of excited levels is used.

LTT = 2: The statistical model is used.

LTT = 3: Energy transfer is described by a transfer matrix.

Card 12 must be given if LTT $\geq$ 1.

CARD	COLUMNS	FORMAT	SYMBOL	
If LTT = 1 or 2 card 12 reads like				
12	1 -11	E11.4	ELCO(1)	
	.	.	.	ELCO(I),I = 1,NU
	.	.	.	
	.	.	.	
	56-66	E11.4	ELCO(6)	

a) For LTT = 1 and NU $\geq$ 2 (excited level description) this means:

ELCO(1): Δ E for first energy level (Δ E is negative for down scattering)

ELCO(2): Partial probability for first level

ELCO(3): Δ E for second level and so on until NU values have been read in. The partial probabilities are normalized by the program and the sum does not have to agree with the total inelastic cross section.

Since in most calculations discrete energy values appear during the course of a history it can happen that in some energy group the subtraction of ΔE leads to a negative value of the energy after the collision. If $E - \Delta E < 0$, for $E_{i+1} \leq E \leq E_i$, the program chooses another level (if possible) or an elastic collision process. In the case where a partial cross section for level excitation does not exist over the whole range of a group, it should only be averaged over the range where it is different from zero:

$$\overline{\sigma_{lev}} = \int_{e}^{E_{i+1}} \sigma_{lev}(E)\, dE \cdot \frac{1}{(E_{i+1} - e)} ,$$

$$e = \begin{cases} E_i , & E_i \geq \Delta E , \\ \Delta E , & E_i < \Delta E . \end{cases}$$

CARD	COLUMNS	FORMAT	SYMBOL

b) For LTT = 2 and NU = 1 (statistical model) ELCO(1) is the nuclear temperature for the group under consideration.

c) If LTT = 3 and NU $\geq$ 3 (transfer matrix version in this energy group)

12	1 - 6	I6	
	7 -17	E11.6	
	18-28	E11.6	
	29-34	I6	ELCO(I),I = 1,NU
	35-45	E11.6	
	46-56	E11.6	

ELCO(1): is the number of energy groups which the neutron skips after the collision.

ELCO(1) = 0: scattering into the same energy group

ELCO(1) = -0: only down scattering in the same group

ELCO(1) = +m (-m): up (down) scattering into a group defined by adding (subtracting) m to (from) the actual energy group index.

ELCO(2) = relative probability for the particle to jump into the energy group specified by ELCO(1).

ELCO(3) = averaged value of the angular distribution (cos θ) in the lab. system for the corresponding scattering process.

ELCO(4) = like ELCO(1) and so on.

Usually the sum of the ELCO(2) + ELCO(5) + ... coincides with the corresponding value of σ_{IN} on card 8.

If this is not the case, i.e. if $\sigma_{IN} \neq \sum$ ELCO(2+3j), the ratio $\sum$ELCO(2+3j)/ σ_{IN} is calculated and during the execution of the Monte Carlo calculations the neutron weight after an inelastic collision is each time multiplied by this factor. This feature can, for example, be used to describe particle multiplication processes like the (n,2n) reaction. For further details see the above paragraph dealing with transfer matrix calculations.

Note that in each energy group another type of inelastic scattering may be used.

Cards 4 to 12 have to be repeated for the next isotope. An End of File card terminates the input of the last isotope.

8.1.2 Output

All the input data are printed by the "LTW Routine" in the same or in a similar format as the input.

8.1.3 Error Messages

The following error messages are possible if there is an inconsistency encountered in the nuclear data input:

FILE NO. (I2) NOT ON DATA INPUT TAPE

If INTAP $\neq$ 5 this message can appear. It signals that the specified file (INFIL) does not exist on the Update Tape of tape unit no. 11 (A6).

FILE NO. (I2) NOT ON SYSTEMS-TYPE (TP.N.(I2))

The library cannot be written on the systems tape (tape unit 9 = A5) since there do not exist the preceeding files with numbers $<$ NFILSY.

ERROR IN THE INPUT OF FISS. SPECTRUM PARAMETERS, MAT = k

Type of error: wrong isotope specification on card MAT = k. (Block I, card 6)

ERROR, ATOMIC-WEIGHT-INPUT. MAT = k

Error in the input of atomic weight etc. Type of error: wrong isotope specification. (Block I, card 7)

ERROR, CROM-INPUT. * MAT = k

Error in the input of microscopic cross section data. Type of error: wrong isotope specification, MAT = k (Block II, card 8)

ERROR, CROM-INPUT, WRONG ENCH, MAT = k, ENE(I) = E

Error in the input of microscopic cross section data. Type of error: wrong energy group on card MAT = k, ENCH = E. (Block II, card 8)

ERROR, INPUT ELAST. SCATT.ANGL. DISTR., WRONG MAT, MAT = k

Error in the input of the distribution function for anisotropic elastic scattering. Type of error: wrong isotope specification, MAT = k (Block III, card 9)

ERROR, INPUT ELAST. SCATT. ANGL. DISTR., WRONG ENCH, MAT = k, ENE(I) = E
Error in the input of the distribution function for anisotropic elastic scattering. Type of error: wrong energy group specification on the card MAT = k, ENCH = E (Block III, card 9)

ERROR, INPUT INEL. SCATT., WRONG ENCH, MAT = k ENE(I) = E
Error in the input of the inelastic scattering model. Type of error: wrong energy group specification on card MAT = k, ENCH = E (Block IV, card 11).

If an error is encountered, no further data are written on the System tape. The job is, terminated, however, by an End of File mark on the System tape in order to enable the computer to put other files behind the incomplete library.

8.2 The Program Tape Writing Routine

8.2.1 Input

CARD	COLUMNS	FORMAT	SYMBOL
1	1 - 6	I6	NFILSY
	7 -12	I6	UNLOAD

The Program Tape Writing Routine writes either the "NUCLEAR DATA PREPARATION (= NDP) Program" , the "RANDOM WALK SAMPLING (RWS) Program" or the "TIME TAPE SCANNING (= TTS) Program" on the specified file number NFILSY of the System Tape (tape unit 9 = A5) (where NFILSY $\leq$ 9).

If UNLOAD is different from zero, the system tape A5 is unloaded after the execution of the present program.

8.2.2 Output

The output messages are:

TIMOC NDP PROGRAM IS WRITTEN ON TAPE A5, FILE (I2)

or

TIMOC RWS PROGRAM IS WRITTEN ON TAPE A5, FILE (I2),

GEOM = (A6): contains the geometry label.

or

TIMOC TTS PROGRAM IS WRITTEN ON TAPE A5, FILE (I2)

8.2.3 Error Messages

The following error messages are possible:

FILE NOT ON TAPE: The specified file cannot be found on the System Tape (tape unit 9 = A5) since the preceeding file(s) do not exist.

RTT ON INPUT TAPE: The monitor input tape (5 = A2) has a redundancy when reading the card containing NFILSY. In this case the "Program Tape Writing Routine" ignores the job.

RTT ON SYST.-TYPE: On the specified file of the System Tape no writing is possible because of unsuccessful redundancy tests. However, an End of File mark is written.

NO INPUT DATA: The input card with NFILSY is missing.

lNCORRECT FILE: The file number NFILSY is 0 or $>$ 9. The writing is ignored.

After writing the "LIBRARY", the "NUCLEAR DATE PREPARATION", the "RANDOM WALK SAMPLING" and the "TIME TAPE SCANNING" Program on the System Tape 9 (= A5) the calculation of a sample problem can be performed. For this purpose the following chain jobs have to be executed.

9. THE NUCLEAR DATA PREPARATION PROGRAM

Contrary to the preparation of the System Tape, the NUCLEAR DATA PREPARATION and the RANDOM WALK SAMPLING Programs have to be executed in the order described here. The different chain programs on the System Tape (NDP, RWS, TTS) are read into the computer each time by a so-called LOADER program. All three chain programs - NDP, RWS and TTS can be read by the same LOADER. Except for the first card specifying the file number on the System tape, the input data for the LOADER are different, depending on what program is loaded into the computer.

9.1 Nuclear Data Input

CARD	COLUMNS	FORMAT	SYMBOL
1	1 - 6	I6	NF

The LOADER reads file NF which corresponds to link 1 (NDP = NUCLEAR DATA PREPARARION Program) from the System Tape A5. $0 < NF \leq 9$

2	1 -60	10A6	TITLE

Title card to identify the job. This job title card is printed as a heading on each page of the output.

3	1 - 4	I4	I: file number specifying the Cross Section Library to be used on the System Tape 9 = A5.
	5 - 8	I4	INR: If $\neq 0$ all microscopic cross sections and nuclear parameters taken from the library

CARD	COLUMNS	FORMAT	SYMBOL
			file are printed in the input.
			If = 0 or blank, no print-out occurs.
4	1 - 4	I4	KM: total number of isotopes to be specified ($\leq$ 20)
	5 - 8	I4	KFM: number of fissionable isotopes
	9 -12	I4	MM: number of mixtures ($\leq$ 20)
	13-16	I4	LM: number of different geometrical regions ($\leq$ 20, LM $\geqslant$ MM)
			In the SORSEC geometry additional regions can be specified in the RWS input. In this case the meaning of LM becomes that of the so-called "material regions" (see (25))
	17-20	I4	NRG: number of the source region. In the SORSEC geometry more than one source region can be specified in the RWS input and the contents of NRG are then ignored.
	21-24	I4	LRG: number of the leakage region. See the LEAK* specification.
5	1 - 6	A6	ISOT
			This set of cards specifies the different isotopes which are used.
			ISOT: isotope identification, one per card. There must be as many cards as there are isotopes specified by KM. Note that fissionable isotopes must be in the beginning.

CARD	COLUMNS	FORMAT	SYMBOL
6	1 - 4	I4	MIXTC: mixture number, in the order 1,2,...M
	5 - 8	I4	NM: number of isotopes in this mixture NM $\leq$ KM
7	1 - 6	A6	ISOT(1)
	7 -18	E12.5	CRO(1)
	.	.	.
	.	.	. ISOT(I), CRO(I),I = 1,NM
	.	.	.
	55-60	A6	ISOT(4)
	61-72	E12.5	CRO(4)
			This card specifies the isotope concentration in the different mixtures.
			ISOT(I): isotope identification
			CRO(I): nuclear density of the isotope (number of atoms in units of 10^{24} per cm^3)
			For each mixture, card 6 and card(s) 7 are required.
8	1 - 4	I4	LCH: region number in increasing order 0,1,2,3,.... LM-1
			This card correlates geometrical region and mixture number.
	5 - 8	I4	JRGN(I): number of the mixture filling this region. There must be a card 8 for each region.

CARD	COLUMNS	FORMAT	SYMBOL
9	1 - 6	I6	IFM: number of different fission spectrum representations to be used (≤ 3)
10	1 - 6	I6	LTT: symbol defining type of fission spectrum (spectrum description in 8.1.1, Block I)
	7 -28	2E11.4	EMIN, EMAX: lower energy limit of the corresponding fission spectrum in eV
	29-61	3E11.4	A,B,C: coefficients of the different fission spectra described in 8.1.1, Block II (C $\neq$ 0 only for LTT = 10)

9.2 The Nuclear Data Output

The NDP program writes the following output:

The job title

The list of isotope specifications and the numbers assigned to them:

ISOTOPE NO.(I2) = (6a)

A list of all mixtures and their isotope composition:

MIXT. NO. (I2)

(A6) - DENS. = (E12.5), (A6) - DENS. = (E12.5),....

A list correlating the region number with its mixture number:

RG(I2) MIXT. NO. (I2)

The parameters of the fission spectra of the primary or source neutrons:

FISS. SPECT. PARAMETERS FOR PRIM. NEUTRONS

FISS. SPECT. *'TYPE NO. (I6), FROM E = (E11.4) TO (E11.4) EV, PARAM.S=(3E11.4)

If the NDP program (link 1) is executed correctly the last message says:
INPUT DATA ARE WRITTEN ON TAPE 10.

If INPR ≠ 0 on card 3, then in addition to the above described output the microscopic cross sections of the specified isotopes are also printed. The arrangement of the nuclear data output corresponds to the input described in 8.1.1. There is only a difference in the sequence of the "Blocks" in the sense that first "Block I" is printed for all isotopes, then "Block II" for all isotopes and so on.

9.3 Error Messages

ERROR, MIXT. INOUT, WRONG SEQ.
The sequence of the isotope mixtures is wrong. Check card 6.

ERROR, MIXT. INPUT
The number of isotopes specified on card 6 is larger than KM on card 4.

ERROR, REGION INPUT
Cards 8 are not in the correct sequence.

ISOTOPE NOT SPECIFIED
An isotope specified in a mixture, card 7 was not specified in the list of cards 5.

RED. CHECK ON LIBRARY TAPE

While reading the Library File on the System Tape the tape check indicator was turned on. In this case either a wrong file number was used and a file containing BCD information was read or the System Tape is physically damaged and has to be replaced and rewritten.

TOO MANY INPUT DATA

The specified nuclear data input exceeds the machine capacity.

ISOTOPE NOT IN LIBRARY

An isotope specification finds no corresponding information in the Library File.

All other error messages refer to a malfunctioning of the computer or eventually to erroneous program changes.

10. THE "RANDOM WALK SAMPLING" PROGRAM

10.1 Operational Modes and Specifications

As already mentioned above, the TIMOC code can be operated in different modes (e.g. Initial value problems = Standard Version, Eigenvalue problems = ITER * option etc.). In addition, different sampling procedures can be applied and a number of quantities calculated optionally. In what follows, the various operational methods and specifications are listed in alphabetical order:

ABSC * : Is meaningful only in connection with the slab and the periodic cylinder geometry. If it is used, each region is subdivided into 10 subregions of equal volume and the absorption in each of these subregions is listed beginning with the innermost subregion and going to the outermost one.

AGE * : The migration area (Fermi Age) is calculated for neutrons slowing down below the lowest energy group or the energy specified by EMIN *.

ANGL * : The direction of the flight vector of the external source neutrons (primary neutrons) is fixed and has to be specified in the input (card 7).

BLOC * : This specification may only be applied, if the sphere geometry is used. It allows to perform calculations in a single unit cell, i.e. a parallelepiped with inscribed concentric spheres. In this case the region around the unit cell gets assigned a region num-

ber which is by definition $RG_{max} + 1$, and has to be defined as the leakage region. The specification BLOC * results then in the neutron not being reflected from the walls of the cell - as it is done in an infinite lattice array - but leaking out of the assembly.

CLLD * : The distribution function of the number of collision is sampled and in the output compared with a Poisson distribution and an equivalent geometrical series.

COMM * : Allows the user of TIMOC to change instructions of the program during the data input (and therefore before execution time). It provides the possibility of making small program changes (card 11).

DPAN * : Can only be used in connection with the TIME * specification. At each boundary crossing an index is written on the Time Tape which specifies the angle interval in which the neutron crosses the boundary. The range of $\cos\Theta$ is divided into 20 equal intervals. Furthermore, in steadystate calculations this angular distribution can only be obtained via the Time Tape.

DUMP * : Gives a dump of all interesting quantities at each collision and each boundary crossing. The same can be achieved by putting sense switch 4 down (see History Dump, Sec. 10.6).

ELP * : The "Expected Leakage Probability" version of TIMOC is executed. In addition (see Sec. 4.4) LEAK * must be specified.

EMIN * : Specifies a lower energy limit which is higher than the lower energy limit of the cross sections in use (card 5)

ENDE * : The energy depositions due to elastic scattering is samplied.

FLUX * : Region and energy dependent fluxes are calculated by the track length estimator.

GRVE * : To each energy interval an average group velocity is assigned. This group velocity is used for calculations of the time parameters instead of the discrete velocities obtained from discrete neutron energies.

ITER * Specifies an eigenvalue calculation where in a multiplying medium successive neutron populations are generated. The sample values are taken from this iteration process under the assumption that the fundamental mode distribution has been reached (see Sec. 6.2).

LEAK * : Specifies a geometrical region as a leakage zone; i.e. neutrons entering this region are assumed to be lost. The leakage region has to be specified in the input.

MTIM * : The mean lifetime for leakage, absorption and slowing down, the mean generation time for each fissionable isotope and the mean scattering time are sampled.

NCOT * : Applies only, when TIME * is specified. The collisions are not registered on the Time Tape.

NCRT * : The same as NCOT * but the boundary crossings are not registered on the Time Tape.

RURU * : The Russian Roulette version is used (Sec. 4.2).

SMEC * : Allows the calculation of differential effects which are smaller than the variance of the sampled quantity itself (Sec. 6.3). The so-called perturbing regions have to be specified in the Geometry Input.

SRRG * : May only be used together with the TIME * option and the SORSEC geometry. If TIME * is specified in the SORSEC geometry only the neutrons crossing the window (see Geometry Description) are registered on the Time Tape. However, specification SRRG * causes all events to be registered. At each crossing or collision point a parameter specifying the sector number is written on the Time Tape. All sector numbers > 9 are set to 0. In the phase of scanning the Time Tape the transmissions and fluxes can be obtained for the different sectors, but not the currents.

STPT * : The source or primary neutrons start from a point source, specified in the input. Card 7 must be specified.

STEN * : The source or primary neutrons start with a fixed energy. Card 7 must be specified.

TIME * : The time dependent solution is desired. At each collision and crossing point all necessary parameters are written on the Time Tape (B6).

TMAX * : Neutron histories are only followed in the time range $0 < t \leq$ TMAX *. Together with the ELP * version this is a powerful variance reducing procedure if time dependent problems are treated. (see Sec. 4.4 and Ref. (10)).

TRAN * : The transmission and the current are calculated at each boundary crossing. In the SORSEC geometry these parameters are only sampled for the window. See also the SRRG * specification.

VARC * : Makes it possible to calculate the probable error of the flux for a specified energy group and region (card 10).

10.2 Input Data and Formats

CARD	COLUMNS	FORMAT	SYMBOL
1	1 - 6	I6	NF NF: LOADER reads file NF which corresponds to Link 2 (= RWS = WANDOMW WALK SAMPLING Program) from the System Tape A5, $0 < NF \leq 9$. If NF = 0, the execution is terminated and all tapes are unloaded.
2	1 -60	10A6	TITLE TITLE: Title card to identify the job. It must agree with card 2 of the NDP input data. The field of columns 61-72 is reserved for the re-run procedure (see Sec. 10.7) and must not be used otherwise.

CARD	COLUMNS	FORMAT	SYMBOL

All following cards are read by the WPKO6 input routine (for description see Ref. (26)) and must therefore be written in the corresponding formats.

3	8 -10	BCD	Columns 8-10 contain the symbols BCD
	12	i	N
			N: number of 6-character-words following in columns 13-72
	13-72	n(a6)	All specifications and options which are used in this computer run. (see Sec. 10.1).
3a	8 -10	TRA	
	12-14	4,4	
4	8 -10	DEC	
	12 cont.	2(i)	HMAX, FISMX
			HMAX: number of primary neutron histories to be calculated.
			FISMX: number of secondary neutron histories to be calculated. (≥ 1)
			When either of these two numbers is reached the calculation is terminated and a full output is printed. For an eventual continuation of the problem see the "Re-Run Procedure" (see Sec. 10.7)
4a	8 -10	TRA	
	12-14	4,4	

CARD	COLUMNS	CONTENTS	SYMBOL
5	8 -10	DEC	Lower energy limit only if EMIN * is specified.
	12 cont.	f	For restrictions see the EMIN * specifications.
5a	8 -10	TRA	
	12-14	4,4	
6	8 -10	DEC	Geometry input data; see Ref. (25)
	12 cont.		
6a	8 -10	TRA	" " " " " "
	12-14	4,4	
7	8 -10	DEC	Only if one or more of the specifications STPT *, ANGL *, STEN * are used.
	12 cont.	f	SPNX, SPNY, SPNZ: x, y, z coordinates of a point source from which the primary neutron histories start if STPT * is specified.
		f	SSINP: Direction coordinates $\sin\varphi$ and
		f	SCOSP: $\cos\varphi$ in the x-y plane, if ANGLE * is specified.
		f	SCOST: Direction coordinate $\cos\vartheta$ (relative to the z-axis) at the source point of all primary neutron histories, if ANGL * is specified.

CARD	COLUMNS	CONTENTS	SYMBOL
		f	STEN: energy [eV] of a primary neutron history at the source point, if STEN * is specified. Unspecified quantities have to be replaced by dummies or zeros, e.g. if only STEN is specified, six numbers (or zeros) have to be written before.
7a	8 -10	TRA	
	12-14	4,4	
8	8 -10	DEC	Only if MTIM * or TIME * is specified.
	12 cont.	i	N N: number of intervals of the time distribution function; $N \leq 20$. If N = 0 the calculation of the time distribution function is omitted.
		f	T(N) Boundaries of the time distribution
		f	T(N-1) intervals. (See Sec. 10.4.11). [sec]
		.	. The quantities have to be in decrea-
		.	.
		.	. sing order of magnitude. If T(0) is
		f	T(0) omitted it is assumed to zero. If TMAX * is specified it is set equal to the highest interval boundary, i.e. TMAX = T(N).

CARD	COLUMNS	CONTENTS	SYMBOL
8a	8 -10	TRA	
	12-14	4,4	
9	8 -10	DEC	Only if GRVE * is specified.
	12 cont.	i	N
			N: number of group velocity parameters. It must be equal to the number of energy groups, i.e.: N = IM
		f	T(N) Average neutron velocity in energy group N. $[cm\ sec^{-1}]$
			T(N-1) The quantities have to be in decreasing order of magnitude.
			$T(N) \geqslant T(N-1)...$
			T(1)
9a	8 -10	TRA	
	12-14	4,4	
10	8 -10	DEC	Only if VARC * is specified.
	12 cont.	i	L: number of geom. or material region
		i	I: number of energy group for which the probable error of the flux is calculated.
10a	8 -10	TRA	
	12-14	4,4	
11	8 -10	OCT	Only if COMM * is specified. There must be as many pairs of cards 11 and 11a as there are instructions to be changed.
	12 cont.	i	I: is the octal address of the instruction to be replaced. It is the address in the listing and is relocated in the computer.

CARD	COLUMNS	CONTENTS	SYMBOL
		i	J: Octal representation of the 36 bit string replacing the contents of the address I in the original version of TIMOC. If the changed instruction contains an address, the absolute machine address has to be used. Since in Fortran II Version 2 the main program of RWS is loaded from location 3000 (octal) one has to add this number to all addresses.
11a	8 -10	TRA	If the foregoing card was not the last change,
	12-14	4,4	else: card 11b
11b	8 -10	TRA	
	12-14	3,4	
12	1 - 6	I6	NF $1 \leq NF \leq 9$: LOADER reads file NF which corresponds to a Link 1, Link 2 or Link 3 problem. This means either a new problem with "NUCLEAR DATA PREPARATION" and "RANDOM WALK SAMPLING" or only a new "RANDOM WALK SAMPLING" run is performed. NF = 0 or blank: The execution is terminated and all tapes in use are unload.

10.3 The Intermediate Output

The first line of each page belonging to the output of the Random Walk and Sampling Program contains the name of the code, TIMOC, and the job title specified on the title card (see card 2)

1) It starts with a print out of the Geometry Input Data.

2) After completion of each 1000 primary histories in the Standard Version or of each 1000 secondary neutron histories in the ITER-Version, an intermediate print out of certain integral quantities is performed. These quantities are always cumulative density functions from the beginning of the sampling procedure (i.e. the first history). They are:

   a) The Number of Histories (see 10.4.2)
   b) The Leakage (see 10.4.3)
   c) The Slowing Down Density (see 10.4.4)
   d) The Multiplication Factor (10.4.7)
   e) The Mean Production Time (see 10.4.8)
   f) The Mean Destruction Times (see 10.4.9)
   g) The Mean Number of Collisions (see 10.4.10)

The definition of these quantities can be found in the following chapter (10.4).

At the same time as this intermediate print-out is made, the whole content of the computer is written on tape B5. This makes it possible to continue the computer run if for any reason (interruption, machine error etc.) it does not reach completion. For the "Re-Run Procedure" see Chapter 10.7.

10.4 The Final Output of Sample Values

10.4.1 General Remarks

A full output of all quantities sampled during the execution of the Random Walk Sampling (RWS) Program is performed when the maximum number of primary or secondary histories specified in the input is reached (card 4). It can also be achieved by pressing Sense Switch 5.

The print out begins with the following general information specifying the problem which has been calculated:

a) Job Title Card

b) Specifications and options used in this computer run

c) Geometry Input Data

d) The Source Region or the Source Point, if specified

e) The Free Storage Locations (Printout: THE MACHINE IS FREE FROM ... TO ...

As described in Chapter 6 the code can operate in two different modes, treating initial value problems or performing eigenvalue calculations by the source iteration procedure (see Chapter 2). The sampling formulae for these two operation modes are described separately under the headings: A) "Standard Version" and B) "ITER * Version".

Furthermore the sampling procedure changes if the Expected Leakage Probability method is used. The corresponding sampling formulae are listed under C) "ELP * Option".

The ELP * option can be applied to both the Standard and the ITER * Version. The formulae are only quoted for the Standard Version. The changes for the

ITER * Version are analogous to the difference between Standard Version and ITER * Version if the ELP * option is not used.

The normalization of the results is different for the Standard Version and for Eigenvalue Calculations (ITER * Version). In the Standard Version the sampling results, except for k_{eff}, τ_{PR} and τ_{DE}, are normalized by the number of source neutrons introduced into the system. This is obvious in the case of non multiplying media. In the presence of fissionable materials the sampled quantities (e.g. leakage, absorption etc.) have logically to be defined also as reaction rates per incident neutron. The variance calculations require however that the averages are calculated over all scored histories, i.e. primaries and secondaries. In the code one averages therefore first by $(H+N)W_o$ and multiplies afterwards the reaction rates by $s = \frac{(H+N)}{H}$. For most of the integral quantities such as reaction rates "Q", k_{eff}, mean times etc. the probable error (P.E.=) is sampled. According to definition it is the 50% confidence interval.

The normalization factors of the following paragraphs are listed here and defined by:

$$s = (H+N)/H \tag{10.1a}$$

$$S_{AB} = \sum_{h,j} W_{h,j-1} P_{AB} + \sum_{n,j} W_{n,j-1} P_{AB} \tag{10.2a}$$

$$S_{LE} = \sum_{h} W_{h,J}^{LE} + \sum_{n} W_{n,J}^{LE} \tag{10.3a}$$

$$S_{SD} = \sum_{h} W_{h,J}^{SD} + \sum_{n} W_{n,J}^{SD} \qquad (10.4a)$$

$$S_{PR} = \sum_{h,j} W_{h,j-1} P_{PR} + \sum_{n,j} W_{n,j-1} P_{PR} \qquad (10.5a)$$

and

$$S_{AB} + S_{LE} + S_{SD} = (H+N) W_o \varrho = S \qquad (10.6a)$$

In the ITER * Version (Eigenvalue Calculations) the summation over h is omitted, i.e.

$$S_{AB}^{IT} = \sum_{n} W_{n,j-1} P_{AB} \ (10.2b), \quad S^{IT} = N W_o \varrho \qquad (10.6b)$$

In the Expected Leakage (ELP*) Option the normalization factors become:

$$S_{AB}^{ELP} = \sum_{h,j} W_{h,j}^{*} P_{AB} + \sum_{n,j} W_{n,j}^{*} P_{AB} \qquad (10.2c)$$

$$S_{LE}^{ELP} = \sum_{h,j} W_{h,j}^{*LE} + \sum_{n,j} W_{n,j}^{*LE} \qquad (10.3c)$$

$$S_{SD}^{ELP} = \sum_{h.} W_{h,J}^{*SD} + \sum_{n} W_{n}^{*SD} \qquad (10.4c)$$

$$S^{ELP} = S \qquad (10.6c)$$

Some variance calculations which are shown in the following chapter require that $\sum_h W_o = H$.

10.4.2 Number of Histories

The program distinguishes between primary and secondary neutron histories and prints the following quantities: NUMBER OF PRIMARY HISTORIES (= H), GEN. SECON. $\left[= G(G_{SMEC}) \right]$, CALC. SECOND. $\left[= N(N_{SMEC}) \right]$. The "number of primary histories" is the number of source neutrons introduced into the system. The "generated secondaries" is the number of neutrons produced by primary and secondary neutrons (in a multiplying media) and the "calculated secondaries" is the number of actually calculated secondary neutron histories.

If the ITER * Version is used, the "number of primary neutrons" is of the order of 200 for a system around criticality. In this case primary neutrons are injected into the system until 200 secondaries are produced and stored in the buffer storage (Sec. 6.2).

Users should also notice that in the ITER * Version the number of "GEN. SECOND". is about 100-300 higher than the "CALC. SECOND." These are the neutrons "waiting" in the buffer at the moment when the calculation is terminated. It is also important to remember that the ratio of "GEN. SECOND." to "CALC. SECOND". is <u>not</u> an estimate of the multiplication factor, as can be seen from Sec. 6.2.

The numbers (G_{SMEC} and N_{SMEC}) in the brackets behind "GEN. SECOND". and "CALC. SEC". refer to the SMEC * option of TIMOC, i.e. the calculation of small effects. If the region specified by SMEC * contains fissionable materials the number in the brackets records the neutron histories generated (or calculated) in this region for which a perturbation effect is calculated.

10.4.3 Leakage (optional, by LEAK *)

Associated with this quantity is a specific leakage region designated as such by the input. When a neutron first enters this region, its current weight is recorded and the history terminated. Two values are sampled and appear in the print-out: TOTAL LEAKAGE = Q_{LE} and UNCOLLi. LEAKAGE = Q^{o}_{LE} (uncollided leakage), followed by their probable errors in the next line. The print out is omitted for a zero leakage rate.

A) "Standard Version": the leakage rates (normalized by the number of primary neutrons H) Q_{LE} are determined by

$$Q_{LE} = \left(\sum_{h} W^{LE}_{h,J} + \sum_{n} W^{LE}_{n,J} \right) S^{-1}_{S} \qquad (10.7a)$$

where $W^{LE}_{h,J}$ ($W^{LE}_{n,J}$) = neutron weight of a primary (secondary) history after the last collision J at the moment of entering the leakage region.

The probable errors which follow in the next line are calculated by:

$$P.E. = \frac{c}{\sqrt{H}} \left\{ \left[\sum_{h} \left(W^{LE}_{h,J} \right)^2 + \sum_{n} \left(W_{n,J} \right)^2 \right] S^{-1} - \left(\frac{Q_{LE}}{S} \right)^2 \right\}^{\frac{1}{2}} \qquad (10.8a)$$

UNCOLLI. LEAKAGE is calculated for J = 0, i.e. for neutrons which leave the assembly without suffering a collision ($W^{LE}_{h,J} = W^{LE}_{h,0}$ and $W^{LE}_{n,J} = W^{LE}_{n,0}$).

B) ITER * Version: The leakage rate is given by

$$Q_{LE}^{IT} = \sum_{n} W_{n,J}^{LE} S_{IT}^{-1} \qquad (10.7b)$$

and

$$P.E. = \frac{c}{\sqrt{N}} \left\{ \left[\sum_{n} \left(W_{n,J} \right)^2 \right] S_{IT}^{-1} - \left(Q_{LE}^{IT} \right)^2 \right\}^{\frac{1}{2}} \qquad (10.8b)$$

The uncollided leakage is calculated for J = 0.

C) ELP * Option: each collision process contributes to the leakage rate Q_{LE}^{ELP} and the corresponding formulae read like this

$$Q_{LE}^{ELP} = \left(\sum_{h,j} W_{h,j}^{*LE} + \sum_{n,j} W_{n,j}^{*LE} \right) S^{-1} S \qquad (10.7c)$$

and

$$P.E. = \frac{c}{\sqrt{H}} \left\{ \left[\sum_{h} \left(\sum_{j} W_{h,j}^{*LE} \right)^2 + \sum_{n} \left(\sum_{j} W_{n,j}^{*LE} \right)^2 \right] S^{-1} - \left(\frac{Q_{LE}^{ELP}}{S} \right)^2 \right\}^{\frac{1}{2}} \qquad (10.8c)$$

10.4.4 Slowing Down Density

The slowing down density is determined whenever the neutron energy takes a value below the lowest energy group or the energy limit specified by the EMIN * option. In the print out it appears as: SLOWING DOWN DENS. = Q_{SD} and the corresponding probable error.

A)C) Standard Version (also in connection with the ELP * Option): the slowing down density Q_{SD} is calculated by

$$Q_{SD} = \left(\sum_{h} W_{h,J}^{SD} + \sum_{n} W_{n,J}^{SD}\right) S^{-1} s \qquad (10.9a)$$

and

$$P.E. = \frac{c}{\sqrt{H}} \left\{ \left[\sum_{h} \left(W_{h,J}^{SD}\right)^2 + \sum_{n} \left(W_{n,J}^{SD}\right)^2 \right] S^{-1} - \left(\frac{Q_{SD}}{S}\right)^2 \right\}^{\frac{1}{2}} \qquad (10.10a)$$

The SLOWING DOWN DENS. is equivalent to the EPSILON factor in the MOCA codes. There it is defined as a generalized fast fission factor. For further details see (1), (2) and (3).

B) ITER * Version: the slowing down density is sampled by

$$Q_{SD}^{IT} = \sum_{n} W_{n,J}^{SD} \Big/ S^{IT} \qquad (10.9b)$$

and

$$P.E. = \frac{C}{\sqrt{N}} \left\{ \sum_{n} \left(W_{n,J}^{SD}\right)^2 \Big/ S^{IT} - \left(Q_{SD}^{IT}\right)^2 \right\}^{\frac{1}{2}} \qquad (10.10b)$$

10.4.5 The Fermi Age (optional, by AGE *)

The Fermi Age is calculated from the displacement of a neutron between its origin $r_o(x_o, y_o, z_o)$ and the point $r_{SD}(x_{SD}, y_{SD}, z_{SD})$ where it slows down below a preassigned energy limit E_{min}. In the presence of fissionable materials the migration distance of secondary neutrons is measured from their corresponding primary neutron's birth place $r_{h;o}(x_{h;o}, y_{h;o}, z_{h;o})$.

The preceding results refer to all neutrons acquiring an energy below E_{min}. There is some possible ambiguity if E_{min} is greater than the lower limit of the fission spectrum since neutrons born below E_{min} are given a final weight $W_{h,o} = 1.0$, a displacement $(x_{h,J} - x_{h,o})^2 = 0$ and are included in the above sums.

The print out reads like this:

FERMI AGE = τ , AGE(x) = τ_x , AGE(y) = τ_y , AGE(z) = τ_z

and below:

PROB. ERROR = P.E.(τ), P.E.(τ_x), P.E.(τ_y), P.E.(τ_z)

A) Standard Version: According to the definition in the literature the Fermi Age is calculated by the following sampling procedure for each space coordinate separately:

$$\tau_x = \frac{1}{6}\left[\sum_h W_{h,J}^{SD}\left(x_{h,J}-x_{h,o}\right)^2 + \sum_n W_{n,J}^{SD}\left(x_{n,J}-x_{n,o}\right)^2\right] S_{SD}^{-1} \qquad (10.11a)$$

where $W_{h,J}$ is the weight after the last collision where the energy becomes lower than E_{min}; $x_{h,J}$ is the x-coordinate of the last collision point and $x_{h(n),o}$ is the birth place of the primary ancestor of the n'th secondary neutron.

$$P.E. = \frac{c}{6\sqrt{H+N}}\left\{\left[\sum_h W_{h,J}^{SD}\left(x_{h,J}-x_{h,o}\right)^4 + \sum_n W_{n,J}^{SD}\left(x_{n,J}-x_{n,o}\right)^4\right] S_{SD}^{-1} - \left(6\tau_x\right)^2\right\}^{\frac{1}{2}} \qquad (10.12a)$$

The moments τ_y and τ_z are calculated in the same way and the total Fermi age is the sum of the different space moments, i.e. $\tau = \tau_x + \tau_y + \tau_z$.

B) ITER * Version: the Fermi Age is not calculated.

C) ELP * option: the weight factor $W_{h,J}$ ($W_{n,J}$) is replaced by $W^*_{h,J}$ ($W^*_{n,J}$).

10.4.6 Production to Source Ratio

In the presence of fissionable materials the number of secondaries for each fissionable isotope is sampled separately and printed in the following output statement:

NUCL. NUMB. K = (k) PRODUCTION/SOURCE = Q_{PR}

The identification number (K) coincides with the one in the print out of the Data Preparation Program § 9.

A) Standard Version: the production to source ratio = Q_{PR} is normalized by the number of source neutrons introduced into the system.

$$Q_{PR}(k) = \left[\sum_{h,j} W_{h,j-1} P_{PR}(k) + \sum_{n,j} W_{n,j-1} P_{PR}(k)\right] \Big/ H W_0 \varrho \qquad (10.13a)$$

with

$$P_{PR}(k) = \nu(k)\,\Sigma_{FI}(k) \Big/ \Sigma_{TO}$$

B) ITER * Version:

$$Q^{IT}_{PR}(k) = \sum_{n,j} W_{n,j-1} P_{PR}(k) \Big/ S^{-1}_{IT} \qquad (10.13b)$$

The physical meaning of this quantity is the contribution of each fissionable isotope to the multiplication factor. The sum of all production to source ratio values gives the multiplication factor (k^{coll}_{eff}).

. 10.4.7 The Multiplication Factor

For the determination of the multiplication factor both the track length estimator and the collision estimator are used and the results are combined by an error minimizing procedure as described in Sec. 4.6.

A) Standard Version: the dynamic multiplication factor k_{dyn} is sampled and the following pring out is listed:

K(EFF.-COLL.) = k^{coll}_{dyn}, K(EFF.-FLUX) = k^{flux}_{dyn}, K(EFF.-DYN.) = k_{dyn}

The sampling formula for the multiplication factor obtained from collisions is:

$$k_{dyn}^{coll} = \left[\sum_{h,j} W_{h,j-1} P_{PR} + \sum_{n,j} W_{n,j-1} P_{PR}\right] S^{-1} \qquad (10.14a)$$

and

$$P.E. = \frac{C}{\sqrt{H+N}} \left[Var\left(k_{dyn}^{coll}\right)\right]^{\frac{1}{2}} \qquad (10.15a)$$

with

$$Var\left(k_{dyn}^{coll}\right) = \left\{\sum_{h} \left[\sum_{j} W_{h,j-1} P_{PR}\right]^2 + \sum_{n} \left[\sum_{j} W_{n,j-1} P_{PR}\right]^2\right\} S^{-1} - \left(k_{dyn}^{coll}\right)^2$$

The sampling formula for the multiplication factor obtained from flux calculations is:

$$k_{dyn}^{flux} = \left\{ \sum_{h} \left[\left(\sum_{j} W_{h,j-1} D_{h,j} \right) + W_{h,J} R_{h,J} \right] \sum_{k} \nu(k) \sum_{Fl}(k) + \right.$$

$$\left. + \sum_{n} \left[\left(\sum_{j} W_{n,j-1} D_{n,j} \right) + W_{n,J} R_{n,J} \right] \sum_{k} \nu(k) \sum_{Fl}(k) \right\} S^{-1}$$

(10.16a)

and

$$P.E. = \frac{c}{\sqrt{H+N}} \left[Var\left(k_{dyn}^{flux}\right) \right]^{\frac{1}{2}} \qquad (10.17a)$$

with

$$Var\left(k_{dyn}^{flux}\right) = \left\{ \sum_{h} [.....]^2 + \sum_{n} [.....]^2 \right\} S^{-1} - \left(k_{dyn}^{flux}\right)^2$$

The covariance is obtained from

$$\mathrm{Cov}\left(k_{dyn}^{coll}, k_{dyn}^{flux}\right) = \left\{\sum_{h}\left[\sum_{j} W_{h,j-1} P_{PR}\right]\cdot\left[\left(\sum_{j} W_{h,j-1} D_{h,j}\right) + W_{h,J} R_{h,J}\right]\cdot\right.$$

$$\cdot\sum_{k}\nu(k)\sum\nolimits_{Fl}(k) + \sum_{n}\left[\sum_{j} W_{n,j-1} P_{PR}\right]\cdot\left[\left(\sum_{j} W_{n,j-1} D_{n,j}\right) + W_{n,J} R_{n,J}\right]\cdot$$

$$\left.\cdot\sum_{k}\nu(k)\sum\nolimits_{Fl}(k)\right\} S^{-1} \qquad (10.18a)$$

The minimum error combination of the variances and the covariance leads to:

$$k_{dyn} = \alpha_1 k_{dyn}^{coll} + \alpha_2 k_{dyn}^{flux} \qquad (10.19c)$$

where α_1 and α_2 are determined as described in § 4.6. The same holds for the variance of K(EFF.-DYN.).

B) <u>ITER * Version:</u> the code samples the multiplication factor k_{eff} as defined in Sec. 2.1 and lists in the print out:

K(EFF.-Coll.) = k_{eff}^{coll}, K(EFF.-FLUX) = k_{eff}^{flux}, K(EFFECTIVE) = k_{eff}

$$k_{eff}^{coll} = \sum_{n,j} W_{n,j-1} P_{PR} / S_{IT}$$

$$\mathrm{Var}\left(k_{eff}^{coll}\right) = \sum_{n}\left[\sum_{j} W_{n,j-1} P_{PR}\right]^2 S_{IT}^{-1} - \left(k_{eff}^{coll}\right)^2 \qquad (10.14b)$$

and

$$P.E. = \frac{c}{\sqrt{N}} \left[Var\left(k_{eff}^{coll}\right) \right]^{\frac{1}{2}} \qquad (10.15b)$$

$$k_{eff}^{flux} = \left\{ \sum_{n} \left[\left(\sum_{j} W_{n,j-1} D_{n,j} \right) + W_{n,J} R_{n,J} \right] \sum_{k} \nu(k) \sum_{Fl}(k) \right\} S_{IT}^{-1}$$

$$Var\left(k_{eff}^{flux}\right) = \left\{ \sum_{n} \left[\ldots\ldots \right]^{2} \right\} S_{IT}^{-1} - \left(k_{eff}^{flux}\right)^{2} \qquad (10.16b)$$

$$P.E. = \frac{c}{\sqrt{N}} \left[Var\left(k_{eff}^{flux}\right) \right]^{\frac{1}{2}} \qquad (10.17b)$$

Both sample values of the multiplication factor are combined to

$$K(EFFECTIVE) = \alpha_1 \, k_{eff}^{coll} + \alpha_2 \, k_{eff}^{flux} \qquad (10.19b)$$

where α_1 , α_2 and $Var(k_{eff})$ are calculated as shown in Sec. 4.6.

C) <u>ELP * Option</u>: $W_{h,j-1}$ and $W_{n,j-1}$ of the collision estimator have to be replaced by $W^{*}_{h,j}$ and $W^{*}_{n,j}$ (eq. 4.15). In the flux estimator the term $\left[\left(\sum_{j} W_{h,j-1} D_{h,j} \right) + W_{h,J} R_{h,J} \right]$ has to be replaced by $\left[\left(\sum_{j} W^{*}_{h,j} D_{h,j} + W^{*LE}_{h,j} R_{h,j} \right) \right]$ as shown in Eq. (4.14).

Small Effect Calculation of the Multiplication Factor (Optional, by SMEC ⋆)

If the SMEC ⋆ option is used there appear in the print out additional multiplication factors which are called: MULTIPLICATION FACTOR IF THE SMEC ⋆ REGION IS BLACK: The next line contains the same quantities as described above but calculated under the condition that all neutron histories after touching the specified region are treated as perturbed histories and do not contribute to the multiplication factor of this (second) print out. The perturbed region is specified by a number in the geometry input. The SMEC ⋆ option is applicable to the GSPHER, the SOR3EC and O5R geometries.

As described in the section on "Calculation Schemes" the purpose of the SMEC ⋆ option is the calculation of differential effects such as, for example, the reactivity value of a small part (control rod, safety plug etc.) of the assembly.

Usually one has to perform two calculations, one where the region of interest ist filled with the material under investigation and a second where the region is empty or replaced by another material (see example 2 in the annex). In order to determine the differential effect one has always to compare output values for the same number of secondary neutrons. Also the "LAST USED RANDOM NUMBER" must coincide in both cases. This is a necessary but not sufficient condition to gurantee that both computer runs use the same histories. If the region of interest is behind the outer convex periphery of the assembly only one calculation is necessary to determine the reactivity value of this region. In this case the second multiplication factor already gives the value for the unperturbed assembly. (See example No. 3 in the annex). If the perturbed region contains a fissionable isotope(s), it has to get (an) isotope name(s) which is (are) different from the other isotope names of fissionable materials even if itis physically the same material as used in other parts of the core.

10.4.8 The Mean Production Time (optional; by MTIM * or TIME *)

The different mean generation times are all calculated as the first moments (Eq. (2.21) and (4.29)) from the neutron's birth to its disappearance, regardless of whether ITER * is specified or not. The Mean Production Time $\tau_{PR}(k)$ of the k's isotope appears in the output statement: NUCL. NUMB. K = k, MEAN GEN. TIME = $\tau_{PR}(k)$.

A) τ_{PR} "Standard Version": the mean production time of isotope k is sampled by:

$$\tau_{PR}(k) = \left[\sum_{h,j} W_{h,j-1}\, t_{h,j}\, P_{PR}(k) + \sum_{n,j} W_{n,j-1}\, t_{n,j}\, P_{PR}(k)\right] S_{PR}^{-1} \qquad (10.20a)$$

where the identification number k of the isotope coincides with that given in the print out of the Data Preparation Program. $P_{PR}(k)$ is the fission probability of the k's isotope, which is usually energy and space dependent and therefore in a certain sense a function of the collision number j $\left[P_{PR}(k) = \nu(k)\Sigma_{FI}(k)/\Sigma_{TO}\right]$. $t_{h,j}$ and $t_{n,j}$ is the time from the birth to the collision point for a primary or secondary neutron, respectively. The probable error follows directly the print out of $\tau_{PR}(k)$

$$P.E. = \frac{c}{\sqrt{H+N}} \left\{ \left[\sum_{h,j} W_{h,j-1} t_{h,j}^2 P_{PR}(k) + \sum_{n,j} W_{n,j-1} t_{n,j}^2 P_{PR}(k) \right] S^{-1} - \left[\tau_{PR}(k) \right]^2 \right\}^{\frac{1}{2}} \qquad (10.21a)$$

The weighted sum of the partial generation times of the different isotopes is the MEAN GENERATION TIME = τ_{PR} of the total system. It is calculated in the same way, only $P_{PR}^{(k)}$ is replaced by the sum $\sum_k P_{PR}(k)$.

B) ITER * Version:

$$\tau_{PR}^{IT}(k) = \sum_{n,j} W_{n,j-1} t_{n,j} P_{PR}(k) / S_{IT} \qquad (10.20b)$$

and

$$P.E. = \frac{c}{\sqrt{N}} \left\{ \left[\sum_{n,j} W_{n,j-1} t_{n,j}^2 P_{PR}(k) \right] S_{IT}^{-1} - \left[\tau_{PR}^{IT}(k) \right]^2 \right\}^{\frac{1}{2}}$$

(10.21b)

Again the weighted sum of the partical generation times is the MEAN GENERATION TIME = τ_{PR} of the total system as already mentioned above. The physical meaning of this quantity was already discussed in Chapter 2. It is, roughly speaking, the mean time between successive neutron generations.

C) ELP * Version: $W_{h,j-1}$, ($W_{n,j-1}$) of the collision estimator has to be replaced by $W^{*}_{h,j}$ ($W^{*}_{n,j}$) of Eq. (4.15).

10.4.9 The Mean Destruction (Absorption and Leakage) Time

(optional, by MTIM * or TIME *)

The mean destruction time is defined as the mean time a neutron stays in the reactor before it is either absorbed or lost by leakage or if EMIN * is specified, by slowing down below the lower energy boundary. The following mean destruction times are listed in the output:

MEAN DESTR. TI = τ_{DE}, MEAN ABSOR. TI = τ_{AB}

MEAN LEAK. TI = τ_{LE}, SLOWING DOWN T. = τ_{SD}

A) Standard Version: the different partial destruction times are calculated in the following way (see Eq. (4.31)):

$$\text{MEAN ABSORPTION TIME } \tau_{AB} = \left[\sum_{h,j} W_{h,j-1}\, t_{h,j}\, P_{AB} + \sum_{n,j} W_{n,j-1}\, t_{n,j}\, P_{AB}\right] S^{-1}_{AB} \qquad (10.22a)$$

where $t_{h,j} = \sum_{g=1}^{j} \frac{D_{h,g}}{C\sqrt{E_g}}$ and $D_{h,g}$ the distance between collision points,

and

$$P.E.(\tau_{AB}) = \frac{c}{\sqrt{H+N}}\left[\left(\sum_{h,j} t_{h,j}^2 W_{h,j-1} P_{AB} + \right.\right. \tag{10.23a}$$

$$\left.\left. + \sum_{n,j} t_{n,j}^2 W_{n,j-1} P_{AB}\right) S_{AB}^{-1} - \tau_{AB}^2\right]^{\frac{1}{2}}.$$

MEAN LEAKAGE TIME τ_{LE} =

$$\left(\sum_{h} t_{h,J}^{LE} W_{h,J}^{LE} + \sum_{n} t_{n,J}^{LE} W_{n,J}^{LE}\right) S_{LE}^{-1} \tag{10.24a}$$

and $t_{h,J}^{LE}$ is the time at which the neutron, after its last collision J, leaks out of the assembly with weight $W_{h,J}^{LE}$:

$$t_{h,J}^{LE} = \sum_{j=1}^{J} \frac{D_{h,j}}{C\sqrt{E_j}} + \frac{R_{h,j}}{C\sqrt{E_j}}$$

where $R_{h,J}$ is the geometrical distance from the last collision point J to the outer boundary along the flight vector.

$$P.E.(\tau_{LE}) = \frac{c}{\sqrt{H+N}}\left\{\left[\sum_{h}\left(t_{h,J}^{LE}\right)^2 W_{h,J}^{LE} + \right.\right.$$

$$\left.\left. + \sum_{n}\left(t_{n,J}^{LE}\right)^2 W_{n,J}^{LE}\right] S_{LE}^{-1} - \tau_{LE}^2\right\}^{\frac{1}{2}}. \tag{10.25a}$$

MEAN SLOWING DOWN TIME τ_{SD} =

$$\left(\sum_{h} t_{h,J}^{SD} W_{h,J}^{SD} + \sum_{n} t_{n,J}^{SD} W_{n,J}^{SD}\right) S_{SD}^{-1} \qquad (10.26a)$$

$t_{h,J}$ is the time to the last collision point, where the neutron energy falls below the lower threshold.

$$P.E.(\tau_{SD}) = \frac{c}{\sqrt{H+N}} \left\{ \left[\sum_{h} \left(t_{h,J}^{SD}\right)^2 W_{h,J}^{SD} + \sum_{n} \left(t_{n,J}^{SD}\right)^2 W_{n,J}^{SD} \right] - \tau_{SD}^2 \right\}^{\frac{1}{2}}. \qquad (10.27a)$$

MEAN DESTRUCTION TIME τ_{DE} =

$$\left\{ \sum_{h} \left[\left(\sum_{j} t_{h,j} W_{h,j-1} P_{AB} \right) + t_{h,J} W_{h,J} \right] + \sum_{n} \left[\left(\sum_{j} t_{n,j} W_{n,j-1} P_{AB} \right) + t_{n,J} W_{n,J} \right] \right\} S^{-1} \qquad (10.27a)$$

where $t_{h,J}\ W_{h,J} = t_{h,J}^{LE}\ W_{h,J}^{LE}$ or $t_{h,J}^{SD}\ W_{h,J}^{SD}$.

The Mean Destruction Time is the weighted sum of all the partial mean values, i.e.

$$P.E.(\tau_{DE}) = \frac{c}{\sqrt{H+N}} \left\{ \sum_{h} \left[\left(\sum_{j} t_{h,j}^2 W_{h,j-1} P_{AB} \right) + t_{h,J}^2 W_{h,J} \right] S^{-1} + \right.$$

$$\left. + \sum_{n} \left[\left(\sum_{j} t_{n,j}^2 W_{n,j-1} P_{AB} \right) + t_{n,J}^2 W_{n,J} \right] S^{-1} - \tau_{DE}^2 \right\}^{\frac{1}{2}} \qquad (10.29a)$$

IF $\tau_{SD} \neq 0$, the destruction time has only a restricted physical meaning since a neutron history is cut at a certain energy threshold, which, from a physical point of view, is an arbitrary operation.

B) ITER * Version:

MEAN DESTRUCTION TIME $\tau_{DE}^{IT} =$

$$\left\{ \sum_{n} \left[\left(\sum_{j} t_{n,j} W_{n,j-1} P_{AB} \right) + t_{n,J} W_{n,J} \right] \right\} S_{IT}^{-1} \qquad (10.28b)$$

and

$$P.E.(\tau_{DE}^{IT}) = \frac{c}{\sqrt{N}} \left\{ \sum_{n} \left[\left(\sum_{j} t_{n,j}^2 W_{n,j-1} P_{AB} \right) + \right. \right.$$

$$\left. \left. + t_{n,J}^2 W_{n,J} \right] \right\} S_{IT}^{-1} - \left(\tau_{DE}^{IT} \right)^2 \Bigg\}^{\frac{1}{2}} . \qquad (10.29b)$$

The sampling of τ_{AB}^{IT}, τ_{LE}^{IT}, τ_{SD}^{IT} and their probable errors is performed in the same way and does not need to be written explicitly.

C) ELP * Option: Eq. (10.28) changes for the mean destruction time to:

$$\tau_{DE}^{ELP} = \left\{ \sum_{h} \left[\left(\sum_{j} t_{h,j}^{*} W_{h,j}^{*} + t_{h,j}^{\prime} W_{h,j}^{*LE} \right) + t_{h,J}^{SD} W_{h,J}^{*} \right] + \sum_{n} \left[\left(\sum_{j} t_{n,j}^{*} W_{n,j}^{*} + t_{n,j}^{\prime} W_{n,j}^{*LE} \right) + t_{n,J}^{SD} W_{n,J}^{*} \right] \right\} S^{-1}$$

(10.28c)

and

$$P.E.(\tau_{DE}^{ELP}) = \frac{C}{\sqrt{H+N}} \left\{ \sum_{h} \left[\left(\sum_{j} t_{h,j}^{*2} W_{h,j}^{*} + t_{h,j}^{\prime 2} W_{h,j}^{*LE} \right) + \left(t_{h,J}^{SD} \right)^{2} W_{h,J}^{*} \right] S^{-1} + \sum_{n} [\ldots\ldots] S^{-1} - \left(\tau_{DE}^{ELP} \right)^{2} \right\}^{\frac{1}{2}}.$$

(10.29c)

The sampling procedure for the partial destruction times τ_{AB}, τ_{LE} and τ_{SD} follows directly from Eq. (10.22a) to (10.27a).

10.4.10 The Number of Collisions (optional, by CLLD ★)

The print out lists: MEAN NUMBER OF COLLISIONS = Q_{CL} (for the whole system).

I) The Mean Number of Collisions

A_I) Standard Version:

$$Q_{CL} = \left(\sum_h \sum_j W_{h,j-1} P_{SC} + \sum_n \sum_j W_{n,j-1} P_{SC} \right) \cdot S \cdot S^{-1} \tag{10.30a}$$

and

$$P.E.(Q_{CL}) = \frac{c}{\sqrt{H}} \left\{ \left[\sum_h \sum_j \left(W_{h,j-1} P_{SC} \right)^2 + \sum_n \sum_j \left(W_{n,j-1} P_{SC} \right)^2 \cdot \right. \right.$$
$$\left. \left. \cdot S^{-1} - S^{-2} Q_{CL}^2 \right\}^{\frac{1}{2}} \right. . \tag{10.31a}$$

B_I) ITER ★ Version:

$$Q_{CL}^{IT} = \left(\sum_n \sum_j W_{n,j-1} P_{SC} \right) S_{IT}^{-1} \tag{10.30b}$$

and

$$P.E.(Q_{CL}^{IT}) = \frac{C}{\sqrt{N}} \left\{ \left[\sum_{n} \sum_{j} \left(W_{n,j-1} P_{SC} \right) \right]^2 S_{IT}^{-1} - Q_{CL}^2 \right\}^{\frac{1}{2}}. \qquad (10.31b)$$

C_I) ELP * Option: the quantities $W_{h,j-1}$ and $W_{n,j-1}$ are replaced by $W^*_{h,j}$ and $W^*_{n,j}$ resp.

II) The Distribution Function of the Number of Collisions

If the specification CLLD * is used the distribution function of the number of collisions is also given in a table containing the fraction of neutrons undergoing one, two, three etc. collisions. For the purpose of comparison, the corresponding Poisson distribution and the geometrical series having the same mean is listed in the print out. The heading of this list reads like: NB. OF COLL. NEUTRONS/POISSON/GEOM. S. The list contains in each line in consecutive columns: the number of collisions as an integer, the corresponding fraction of neutrons which make this number of collisions during their life (normalized to unity), the Poisson distribution, and the distribution of a geometrical series.

The distribution function is given as the number of neutrons suffering J collisions before removal by leakage, absorption or slowing down. The Poisson distribution and the geometrical series are both normalized to the mean number of collisions provided by the Monte Carlo calculation.

The fraction of neutrons which suffers 100 collisions and more are printed together in one line. The last line contains the sum of all previous quantities. They are unity, except for the last two numbers and there only in the case of more than 100 collisions. The difference between one and the printed sum is just the contribution of terms with an index $n > 100$.

A_{II}) Standard Version:

$$Q_{CL}(j) = \left[\sum_h \left(W_{h,j} - W_{h,j+1}\right) + \sum_n \left(W_{n,j} - W_{n,j+1}\right)\right] \cdot S^{-1} . \qquad (10.32a)$$

The zero collision contribution is calculated separately by the uncollided leakage term.

B_{II}) ITER * Version:

$$Q_{CL}^{IT}(j) = \left[\sum_n \left(W_{n,j} - W_{n,j+1}\right)\right] \cdot S_{IT}^{-1} . \qquad (10.32b)$$

C_{II}) ELP * Option:

The quantities $W_{h,j-1}$ and $W_{n,j-1}$ are replaced by $W^*_{h,j}$ and $W^*_{n,j}$ resp.

10.4.11 Production and Destruction Time Distribution (optional, by MTIM * or TIME *)

The two functions are defined by Eq. (2.19) and Eq. (2.20). In the literature similar functions are described under the name "Distribution of slowing down times to first fission" [23], [24].

In the print out, the heading of these tabulated functions have the following meaning:

TIME INTERVAL $\left[= t_g - t_{g+1}\right]$, PROD.TI*DT $\left[= \int_{t_g}^{t_{g+1}} Q_{PR}(t)\,dt\right]$,

DE.TI.*DT $\left[= \int_{t_g}^{t_{g+1}} Q_{DE}(t)dt\right]$, PR.TI.DI $\left[= (t_{g+1}-t_g)^{-1}\int_{t_g}^{t_{g+1}} Q_{PR}(t)\,dt\right]$,

DE.TI.DI $\left[= (t_{g+1}-t_g)^{-1}\int_{t_g}^{t_{g+1}} Q_{DE}(t)dt\right]$, CO.TO.PR.TI. $\left[= \int_{t_g}^{t_{g+1}} t\,Q_{PR}(t)dt\right]$,

CO.TO.DE.TI. $\left[= \int_{t_g}^{t_{g+1}} t\,Q_{DE}(t)dt\right]$.

Neutrons born with an energy smaller than E_{min} appear only in the normalization factor of the denominator of the following equations.

If the TMAX * option is used, neutrons with a lifetime larger than t_{max} are excluded and are not considered either in the mean values or in the distribution functions.

The time intervals printed in the first column have to be specified in the input (card 8).

A) Standard Version: Columns two and three contain the two distribution functions $Q_{PR}(t)$ and $Q_{DE}(t)$ integrated over the corresponding time interval, i.e.

$$\int_{t_g}^{t_{g+1}} Q_{PR}(t)dt = \left\{ \sum_h \sum_j \left[W_{h,j-1} \right]_g \sum_k P_{PR}(k) + \right.$$

$$\left. + \sum_n \sum_j \left[W_{n,j-1} \right]_g \sum_k P_{PR}(k) \right\} S_{PR}^{-1} \qquad \text{(10.33a)}$$

where

$$\left.\begin{matrix} \left[W_{h,j-1} \right]_g \\ \left[W_{n,j-1} \right]_g \end{matrix}\right\} = \begin{cases} \begin{cases} W_{h,j-1}, \\ W_{n,j-1}, \end{cases} & t_g < t_{h,j} \leq t_{g+1}, \\ 0, & \text{otherwise,} \end{cases}$$

and

$$\int_{t_g}^{t_{g+1}} Q_{DE}(t)dt = \left\{ \left[\sum_h \left(\sum_j W_{h,j-1} P_{AB} \right) + W_{h,J}^{LE} + W_{h,J}^{SD} \right]_g + \right.$$

$$\left. + \left[\sum_n \left(\sum_j W_{n,j-1} P_{AB} \right) + W_{n,J}^{LE} + W_{n,J}^{SD} \right]_g \right\} S^{-1}.$$

(10.34a)

The next two columns contain the mean value of the function $Q_{DE}(t)$ and $Q_{PR}(t)$ in the specified time interval, i.e.

$$\overline{Q_{PR}(t_g)} = \frac{1}{(t_{g+1}-t_g)} \int_{t_g}^{t_{g+1}} Q_{PR}(t)\,dt \qquad (10.35a)$$

and

$$\overline{Q_{DE}(t_g)} = \frac{1}{(t_{g+1}-t_g)} \int_{t_g}^{t_{g+1}} Q_{DE}(t)\,dt \qquad (10.36a)$$

The last two columns contain finally the contribution of the particular time interval to the corresponding mean production and mean destruction time or in other words $\int_{t_g}^{t_{g+1}} t Q_{PR}(t)\,dt$ and $\int_{t_g}^{t_{g+1}} t Q_{DE}(t)\,dt$, i.e. the importance of the interval for the mean value τ_{PR} and τ_{DE}.

$$\int_{t_g}^{t_{g+1}} t Q_{PR}(t)\,dt = \left\{ \sum_h \sum_j t_{h,j} \left[W_{h,j-1} \right]_g \sum_k P_{PR}(k) + \right.$$

$$\left. + \sum_n \sum_j t_{n,j} \left[W_{n,j-1} \right]_g \sum_k P_{PR}(k) \right\} S_{PR}^{-1} \qquad (10.37a)$$

and

$$\int_{t_g}^{t_{g+1}} t\,Q_{DE}(t)\,dt = \left\{ \sum_h \sum_j t_{h,j} \left[W_{h,j-1} \right]_g P_{AB} + t_{h,J} \left[W_{h,J}^{LE} + W_{h,J}^{SD} \right]_g + \right.$$

$$\left. + \sum_n \sum_j t_{n,j} \left[W_{n,j-1} \right]_g P_{AB} + t_{n,J} \left[W_{n,J}^{LE} + W_{n,J}^{SD} \right]_g \right\} S^{-1}. \qquad \text{(10.38a)}$$

B) ITER * Version

$$\int_{t_g}^{t_{g+1}} Q_{PR}^{IT}(t)\,dt = \left\{ \sum_n \sum_j \left[W_{n,j-1} \right]_g \sum_k P_{PR}(k) \right\} \cdot \left(S_{PR}^{IT} \right)^{-1} \qquad \text{(10.33b)}$$

and

$$\int_{t_g}^{t_{g+1}} Q_{DE}^{IT}(t)\,dt = \left\{ \sum_n \sum_j \left[W_{n,j-1} \right]_g P_{AB} + \left[W_{n,J}^{LE} + W_{n,J}^{SD} \right]_g \right\} S_{IT}^{-1} \qquad \text{(10.34b)}$$

The other formulae for the time distribution functions change in the same way as in the Standard Version described above.

C) ELP * Option:

The terms $\left[W_{h,j-1} \right]_g$ and $\left\{ \sum_j \left[W_{h,j-1} \right]_g P_{AB} + \left[W_{h,J}^{LE} + W_{h,J}^{SD} \right]_g \right\}$ of the above equations have to be replaced by: $\left[W_{h,j-1}^{*} \right]_g$ and $\left\{ \sum_j \left[W_{h,j}^{*} + W_{h,j}^{*LE} \right]_g + \left[W_{h,J}^{*SD} \right]_g \right\}$ respectively.

10.4.12 The Fission Rates

If there are fissionable isotopes, the mean number of fissions per initial neutron is printed out separately for all different regions l under the following lable: RG = l, NUCL. NO K = k, FISSIONS PER PRIM. NEUTRON = Q_{FI}. The region number (RG) and the isotope identification (NUCL. NO) coincide with that given in the print out of the NDP programme.

A) Standard Version:

$$Q_{FI} = \left[\sum_h \sum_j W_{h,j-1} P_{FI}(l,k) + \sum_n \sum_j W_{n,j-1} P_{FI}(l,k)\right] \Big/ H W_o \varrho ,$$

$$P_{FI}(l,k) = \frac{\Sigma_{FI}(l,k)}{\Sigma_{TO}(l)} . \qquad (10.35a)$$

The number of fissions, however, is not calculated in the above indicated manner which would require an extra sampling procedure at each collision point. The calculation is actually performed via the energy dependent absorption rate described later on in the manual, i.e.:

$$Q_{FI}(l,k) = \sum_i \frac{Q_{AB}^{flux}(l,i)\, Q_{PR}(l,i,k)}{\nu(l,i,k)\left[1 - Q_{SC}(l,i)\right]} \qquad (10.36a)$$

where $Q_{AB}^{Flux}(l,i)$ is the absorption rate in the energy group i and the geometrical region l.

The physical meaning of the fission rate is simply the number of fissions per primary neutron. The ratio of the multiplication factor k_{dyn}^{flux} and the sum of the fission ratios gives the averaged $\overline{\nu}$-value of the system.

In lattice calculations one can calculate directly the fast fission ratio δ_{28}, δ_{25}, etc. from this quantity by multiplying it by the thermal value ν_{th} of the system under consideration.

B) ITER * Version:

$$Q_{Fl}^{IT} = \left[\sum_{n} \sum_{j} W_{n,j-1} P_{Fl}(l,k)\right] S_{IT}^{-1}. \qquad (10.35b)$$

C) ELP * Option:

$W_{n,j-1}$ $(W_{n,j-1})$ of the above equations is replaced by

$W^{*}_{n,j}$ $(W^{*}_{n,j})$.

10.4.13 Last Used Random Number and Total Number of Records on Time Tape

The quantity: LASED USED RANDOM NUMBER is a print out of the initial random number of the last neutron history. In TIMOC a special subroutine produces for each history a new initial random number which is the basis for a chain of random numbers in the history under consideration. This technique is used to allow the calculation of differential effects. In calculations where no multiplying media are used, this number can be used to continue a calculation and can be fed in by means of the COMM * option. The number is located at 1RNIN (see the corresponding description).

If the TIME * option is used the TOTAL NUMBER OF RECORDS ON TIME TAPE is printed. It gives an indication of the space used on the time tape. A normal tape may contain between 4.000 and 5.000 of these records. One record contains the parameters of 100 events (boundary crossings and collisions). If the number of collisions is $\sum_h J(h) + \sum_n J(n)$, of boundary crossings $\sum_h G(h) + \sum_n G(n)$ and of histories H+N, then

$$\sum_h \left[J(h)+G(h)+1\right] + \sum_n \left[J(n)+G(n)+1\right] = 100\,R.$$

(The starting parameters of a neutron are also subject to the contribution to the time tape). If the NCOT * option is used, this reduces to:

$$\sum_h G(h) + \sum_n G(n) = 100\,R.$$

10.4.14 Energy Dependent Flux and Absorption Rates (optional, by FLUX *)

The print out of the energy dependent flux and absorption rates is done in a table under the heading of RG, ENERGY GROUP, ABSORPTION, ABSORP./VOL., FLUX(E)*DE*VOL, FLUX(E)*DE, FLUX(E), and FLUX(E)*E. In these labels FLUX(E) stands for energy dependent flux, E for energy and DE for $\Delta E = E_{i+1} - E_i$, where the E_i are the boundaries of the different energy groups. The column labeled with RG contains (in integer form) the region number of the spatial region l. This number coincides with that given on the print out of the NPD Programme. The column labeled ENERGY GROUP contains the boundaries E_i in MeV.

FLUX(E)*DE*VOL ($= Q_{Flux}$) is the basic quantity being sampled during the calculation. It is the volume and energy integrated flux calculated by the track length estimator for each region and energy group separately.

A) Standard Version:

$$Q_{Flux}(l,i) = \int_{E_i}^{E_{i+1}} dE \int_{V_l} dV \Phi(\vec{r}, E) =$$

$$= \left\{ \sum_h \left[\sum_j W_{h,j-1}(i) D_{h,j}(l) \right] + W_{h,J}(i) R_{h,J}(l) + \right.$$

$$\left. + \sum_n \left[\sum_j W_{n,j-1}(i) D_{n,j}(l) \right] + W_{n,J}(i) R_{n,J}(l) \right\} \cdot s \cdot S^{-1} \qquad (10.37a)$$

where $D_{h,j}(l,i)$ is that part of the distance between the (j-1)th and the j'th collision which lies in region l and $R_{h,j}(l)$ is that part of the

history track length which lies in region l after the last collision. Usually $R_{h,j}$ is the track length before leakage. $W_{h,j}(i)$ indicates that the summation is only performed over collisions which belong to the energy group i.

B) ITER * Version:

$$Q^{IT}_{Flux}(l,i) = \left\{\sum_{n}\left[\sum W_{n,j-1}(i)\,D_{n,j}(l)\right] + W_{n,J}(i)\,R_{n,J}(l)\right\} S^{-1}_{IT} \,. \qquad (10.37b)$$

C) ELP * Option:

$$Q^{ELP}_{Flux}(l,i) = \left\{\sum_{h}\left[\sum_{j} W^{*}_{h,j}(i)\,D_{h,j}(l) + W^{*LE}_{h,j}(i)\,R_{h,j-1}(l)\right] + \sum_{n}\left[\sum_{j} W^{*}_{n,j}(i)\,D_{n,j}(l) + W^{*LE}_{n,j}(i)\,R_{n,j-1}(l)\right]\right\} \cdot s \cdot S^{-1} \qquad (10.37c)$$

(see Eq. (4.14))

If both ITER * and ELP * are specified, Eq. (10.37) has to be written in the form of Eq. (10.37b).

FLUX(E)*DE is the volume averaged flux integrated over the energy interval $\left[E_j, E_{j+1}\right]$, i.e.:

$$\text{FLUX(E)*DE} = \frac{1}{V_1} \cdot Q_{\text{Flux}} . \tag{10.40}$$

FLUX(E) is the average flux value with respect to energy and geometrical region

$$\text{FLUX(E)} = \frac{1}{V_1 \, (E_{i+1} - E_i)} \, Q_{\text{Flux}} \tag{10.41}$$

where V_1 is the volume of region 1 (in cm^3) and $(E_{i+1} - E_i)$ the energy difference in eV.

FLUX(E)*E is the product of flux and energy (mainly for the purpose of plotting):

$$\text{FLUX(E)*E} = \frac{Q_{\text{Flux}}}{V_1 \, (E_{i+1} - E_i)} \cdot \frac{\left(E_i + E_{i+1}\right)}{2} . \tag{10.42}$$

All other quantities in this table are directly derived from Q_{Flux}.

ABSORPTION: The energy dependent absorption rate is calculated as the mean value for each geometrical region 1 and energy group i by multiplying the flux value $Q_{\text{Flux}}(1,i)$ with the corresponding macroscopic absorption cross section.

$$\text{ABSORPTION} = Q_{\text{Flux}} \cdot \sum\nolimits_{AB}(1,i). \tag{10.38}$$

This value is also printed out per unit volume (cm^3)

$$\text{ABSORP./VO} = Q_{\text{Flux}} \cdot \sum\nolimits_{\text{AB}}(l,i) \cdot V_l^{-1} \qquad (10.39)$$

The track length estimator is used for the fact that it has in many cases a lower variance than the collision estimator.

Error Calculation for a Specified Geometrical Region and Energy Group (optional by FLUX * and VARC *)

The calculation of the variance of the flux (and associated quantities) for all energy groups and geometrical regions would be a very time consuming procedure. Upon specifying VARC * the code provides an estimate of the P.E. in one phase space interval (energy group "i" of region "l"). Contrary to the other error calculations, P.E.$\left[Q_{\text{Flux}}(l,i) \right]$ is given also in percents, a number which holds for all quantities which are linear functions of $Q_{\text{Flux}}(l,i)$.

The print out reads:

PROBABLE ERRORS = (E12.5) %

followed by a line which corresponds to the original heading RG, ENERGY GROUP, ABSORPTION etc. In this line the region number "l" and the energy group boundaries are followed by the error margins of the different quantities, i.e. P.E.$\left[Q_{\text{AB}}(l,i) \right]$, etc.

10.4.15 The Total Absorption

The total absorption is calculated separately by the track length estimator and the collision estimator during each neutron history. The results are combined in the output by minimizing the error. The corresponding formulae are given in Sec. 4.6 (Double Sampling for Multiplication factor and Absorption Calculations). They correspond exactly to the formulae calculating k_{dyn} or k_{eff} (if ITER * is specified) except for the fact that $\nu\Sigma_{FI}$ and P_{PR} are replaced by Σ_{AB} and P_{AB} respectively.

10.4.16 Spatial Fine Structure of the Absorption

If one uses the SLBCYL geometry and the ABSC * option, one can further get the spatial fine structure of the absorption in each region. For this purpose each geometrical region is divided into 10 subregions of equal volume. In each subregion the absorption is summed over the whole energy range. In this case the absorption is determined from the collision estimator (described in detail in the description of the SLBCYL geometry).

10.4.17 Transmission and Current (optional, by TRAN *)

If the TRAN * specification is used energy dependent transmission rates and the currents are calculated through surfaces defined in the geometry programme.

The quantities which appear in the print out are headed by a line, which contains the labels RADIUS, RG, ENERGY GROUP, TRA(IN/OUT), TRA(OUT/IN),

TRA(SM), CUR(IN/OUT), CUR(OUT/IN) and CUR(SM). These bivariate tables give the transmission and current through surfaces as a function of energy group i and surfaces o, where the surfaces are uniquely related to the regions l.

The column labeled with RADIUS contains in the case of the GSPHER geometry the actual radius of the surface crossed by the neutrons. In the case of the SLBCYL geometry it contains the radius of the surfaces in the cylindrical case and the distance from the origin in the slab case.

The column labeled with RG contains in the spherical and cylindrical cases always the higher region number of the two regions on each side of the surface, i.e. RG = 1 means the surface between region 0 and region 1. In the slab case this column contains the lower of the two region numbers with an exception when the boundary lies between the region 0 and the region with the highest possible number (see also the description of these geometries). In the SORSEC geometry only the transmission through the "window" is determined. The printed region numbers refer to these regions which are cut by the window. As the quantity "radius" is in this case meaningless, the floating point representation of the material region number is printed in this column. The same holds for the RDCLST (rod cluster) geometry. Here again the column labeled with radius contains the floating point representation of the material region number. For further details see the geometry description.

The column labeled ENERGY GROUP contains the energy group boundaries (E_i, E_{i+1}).

The transmission is defined as the number of neutrons entering or leaving a geometrical region through its surface within the specified energy group.

TRA(IN/OUT) = Q_{TRA+} is the number of neutrons leaving the specified region through its surface per unit aerea (cm^2).

$W^+_{h,j}(l,i)$: weight factor of a neutron track (belonging to the energy group E_i) crossing the boundary of region l in the "outward" direction, i.e. $\cos\Theta_l > 0$, where $\cos\Theta$ is the cosine of the neutron track with regard to the surface normal.

F_l : surface of region l . In the case of an infinite (twodimensional) cylindrical geometry the surface is replaced by the circumference. In the slab geometry this quantity is set to unity.

TRA(OUT/IN) = Q_{TRA-} gives the neutron transmission into the opposite direction

TRA(SM) is the sum of both quantities, i.e.

$$\text{TRA(SM)} = \text{TRA(IN/OUT)} + \text{TRA(OUT/IN)}$$

The current across boundaries are defined as the projection of the neutron track on the surface normal. The results are split into outgoing CUR(IN/OUT) = Q_{CUR+} for $\cos\Theta_l > 0$ and incoming CUR(OUT/IN) = Q_{CUR-} for $\cos\Theta_l < 0$ current and the net current CUR(SM) = CUR(IN/OUT) - CUR(OUT/IN). At a boundary of leakage region (LEAK *) the transmission and the current occur only in one direction. Usually it is TRA(IN/OUT) and CUR(IN/OUT).

A) Standard Version:

$$Q_{TRA+}(l,i) = \left\{\sum_h \sum_g W^+_{h,g}(l,i) + \sum_n \sum_g W^+_{n,g}(l,i)\right\} \cdot s \cdot S^{-1} \cdot F_l^{-1} \qquad (10.43a)$$

where g = summation index for neutrons crossing the boundary l in energy group i.

$Q_{TRA-}(l,i)$ is calculated for $W^{-}_{h,g}(l,i)$ and

$$Q_{CUR+}(l,i) = \left\{ \sum_{h} \sum_{g} W^{+}_{h,g}(l,i) \cos\Theta_l + \sum_{n} \sum_{g} W^{+}_{n,g}(l,i) \cos\Theta_l \right\} \cdot s \cdot S^{-1} \cdot F_l^{-1}$$

for $\cos\Theta \geq 0$ (10.44a)

Q_{CUR-} is calculated for $W^{-}_{h,g}$ where $\cos\Theta < 0$.

Finally

$$CUR(SM) = CUR(IN/OUT) + CUR(OUT/IN). \quad (10.45a)$$

This quantity becomes negative under the condition that:

$$CUR(IN/OUT) < |CUR(OUT/IN)|$$

B) <u>ITER * Version</u>:

$$Q^{IT}_{TRA+}(l,i) = \left\{ \sum_{n} \sum_{g} W^{+}_{n,g}(l,i) \right\} (S_{IT} F_l)^{-1}. \quad (10.43b)$$

The other quantities Q^{IT}_{TRA-}, Q^{IT}_{CUR+}, Q^{IT}_{CUR-} are calculated in the same manner.

C) ELP * Option:

$$Q^{ELP}_{TRA+}(l,i)=\left\{\sum_{h}\left[\sum_{g} W^{+*}_{h,g}(l,i)+\sum_{g'} W^{+*LE}_{h,g'}(l,i)\right]+\right.$$

$$\left.+\sum_{n}\left[\sum_{g} W^{+*}_{n,g}(l,i)+\sum_{g'} W^{+*LE}_{n,g}(l,i)\right]\right\}\cdot s\cdot S^{-1}\cdot F_{l}^{-1}. \qquad (10.43c)$$

The other quantities are calculated in the same manner.

In a second bivariate table all the above mentioned quantities of transmission and current are for convenience also printed per unit energy $\left[eV\right]$, i.e. for example:

$$\text{TRA(IN/OUT) PER UNIT ENERGY} = \frac{Q_{TRA+}}{(E_{i+1} - E_i)} \qquad (10.46)$$

Unit: n/cm^2 eV.

10.4.18 Energy Deposition

By the use of the ENDE * option the energy deposited throughout the course of a history is kept as a bivariate function of region and energy loss range. At each elastic collision point, j, the quantity $W_{h,j}(E_j - E_{j+1})$ is computed. Here the subscripts j and j+1 mean the neutron energy before and after the elastic collision respectively.

The heading of the energy deposition table reads like this: RG, ENERGY GROUP, EN.-DEPOS., EN.-DEP./VO, ED/(VO*DE).

The first column contains the geometrical region number.

The next two columns contain the boundaries of the energy loss range. It should be stressed that the energy loss ranges used here have a different meaning than the energy bounds used elsewhere, e.g. flux, cross section representation etc. The numerical values of the energy deposition ranges are only for convenience identical with the energy group structure.

The column containing the integrated energy deposition

$$Q_{ED} = \int_{V_i} dV \int_{E_i}^{E_{i+1}} dE\, dE'\, n(E',r) P_{EL}(E',r') E(E',r' \rightarrow E,r)(E'-E) \qquad (10.47)$$

is followed by the energy deposition per unit volume (EN.-DEP./VO) and finally in the last column ED./(VO*DE) per unit volume and energy $\left[\frac{}{eV}\right]$ for the purpose of plotting.

The total energy deposition is printed in the last line. It is the average energy loss per incident neutron by elastic scattering.

The subdivision into energy deposition ranges makes it possible to get quantitative results on the magnitude of the momentum transfer during the elastic slowing down process.

A) Standard Version: the energy deposition is calculated by:

$$Q_{ED} = \left\{ \sum_{h} \sum_{j(el)} W_{h,j} (E_j - E_{j+1}) + \right.$$

$$\left. + \sum_{n} \sum_{j(el)} W_{n,j} (E_j - E_{j+1}) \right\} \cdot s \cdot S^{-1} \qquad (10.48a)$$

$$\text{for} \quad E_i \leq (E_j - E_{j+1}) < E_{i+1}$$

where E_i, E_{i+1} are the boundaries of the corresponding energy deposition range.

B) ITER * Version:

$$Q_{ED}^{IT} = \left\{ \sum_{n} \sum_{j(el)} W_{n,j} (E_j - E_{j+1}) \right\} \cdot S_{IT}^{-1} \qquad (10.48b)$$

$$\text{for} \quad E_i \leq (E_j - E_{j+1}) < E_{i+1} .$$

C) ELP * Option:

$$Q_{ED}^{ELP} = \left\{ \sum_{h} \sum_{j(el)} W_{h,j}^{*} \left(E_j - E_{j+1}\right) + \right. \tag{10.48c}$$

$$\left. + \sum_{n} \sum_{j(el)} W_{n,j}^{*} \left(E_j - E_{j+1}\right) \right\} \cdot s \cdot S^{-1}.$$

10.4.19 Fission spectra

The final print out is headed by a line with the label ENERGY GROUP, No.NTR., NO.COLLS., CHI-PRIM*DE, CHI-SEC*DE, CHI-PRIM and CHI-SEC.

The column labeled with NO.NTR. gives the number of neutrons (primary and secondary together) which are born within the respective energy groups. Note that only those neutrons are counted which have an initial energy above E_{min}.

The column labeled with NO.COLLS. is the total number of collisions encountered by a neutron history as a function of the energy group in which the neutrons were originally generated. It includes both primary and secondary neutrons. But since each collision is counted with the same frequency, i.e. independent of the weight of the neutron, this quantity has no physical meaning. It is however directly related to the machine time required for the calculation. This machine time is roughly composed of three quantities (if we discard the use of the time scanning programme):

$$T = T_O + T_{BC} + T_{CL}$$

T_0 means a constant contribution (independent of the number of histories) which comes from the INPUT programme, tape handling etc. T_{CL} is the machine time required to run through the collision routine. This time is directly proportional to the total number of collisions, which is printed in the last line. T_{BC} is proportional to the total number of boundary crossings, which are made during the calculation.

The column labeled with CHI-PRIM*DE gives the fraction of primary neutrons in the specified energy group and is defined as

$$\text{CHI-PRIM*DE} = \int_{E_i}^{E_{i+1}} \chi(E)\,dE = \frac{1}{H} \sum_{h} W_{h,o}(i). \tag{10.49}$$

Since also here neutrons with an initial energy below E_{min} are not counted, the sum of all CHI-PRIM*DE gives the primary source fraction above E_{min}.

The quantities in the column labeled with CHI-SEC*DE gives the fraction of secondary neutrons in the specified energy group. If there are more than one fissionable isotope in the assembly, the fission spectrum of the secondary neutrons is given by the weighted sum of the single spectra

$$\text{CHI-SEC*DE} = \frac{1}{N} \sum_{n} W_{n,o}(i) \tag{10.50}$$

where $W_{n,o}(i)$ is again the weight of a starting neutron with an energy in the energy group i. Here again neutrons with an initial energy below E_{min} are

excluded and the sum of all CHI-SEC*DE gives the secondary source fraction above E_{min}.

The next two quantities labeled with CHI-PRIM and CHI-SEC are simply obtained by dividing the previous quantities through the group width ΔE_i. The physical meaning of these quantities is the average value of the primary (secondary) fission spectrum $\chi(E)$ in the energy group (E_i, E_{i+1}).

$$CHI-PRIM = \frac{1}{(E_{i+1}-E_i)} \int_{E_{i+1}}^{E_i} \chi(E)dE =$$

$$= \frac{CHI-PRIM*DE}{(E_{i+1}-E_i)}. \qquad (10.51)$$

The same holds for the spectrum of the secondary neutrons.

$$CHI-SEC = \frac{CHI-SEC*DE}{(E_{i+1}-E_i)}.$$

10.5 Error Messages

During execution the programme performs a number of checks and test calculations. They control the proper functioning of the tapes (redundancy etc.), the consistency of the data input and obvious calculational errors.

The error message is composed of the label: TYPE OF ERROR..... LOCATION.... **** followed either by CALC. CONTIN. (calculation continued) or CALL EXIT. The text following the "type of error" is a 6 character BCD word. If its

last character is a 1 the calculation is terminated by calling the exit routine. The location is an octal integer refering to the storage location of the error stop in the listing of the RWS programme.

The different error messages are listed in the following and explained.

EMIN 1 — The energy cut off E_{min} (EMIN * option) is lower than the lowest boundary of the energy group structure of the cross section set in use.

EOF-S1 — EOF on tape S (B5): Tape B5 contains an End of File instead of information.

FFICT 1
FIWCT 1 — Either an undetected error in the cross section set or machine error when writing or reading tape B5. The best is first to run the same case again without making any change.

GV-IM1 — The number of group velocities (GRVE * option) does not coincide with the number of energy groups (Sec. 10.2 card 9).

GV-WO1 — The group velocities (Sect. 10.2 card 9) are not in the right (decreasing) order.

LSTMD1 — There are more than 20 time groups for the lifetime and generation time distribution (Sect. 10.2 card 8) (Storage overflow of LTSP BES21).

LS-WO1 — The boundaries for the lifetime and generation time spectrum are not in the right (decreasing) order (Sect. 10.2 card 8).

NOFRT 1 This message may appear, if the TIME * option is used and a rerun is made (rerun see Sect. Lo.2). It signals that the time tape is not in the proper condition, i.e. either the tape is destroyed or a wrong tape is used.

Q-WRW Attention! A specification is detected which is not contained in the list in the programme. This specification is omitted.

RTT-S1 Redundancy on tape S(B5): Tape B5 contains a redundant portion, i.e. it is destroyed and has to be replaced.

RTT-W Attention! Redundancy when writing the contents of the storage on tape B5. The tape is filled up to the redundant part with small records, which contain no information. Afterwards the information is correctly written. Since these small records can be detected, no information is lost, because the programme is able to space the tape in the proper position when reading tape B5 for a rerun.

RT-WU Attention! Redundancy when writing time dependent information on tape B6. The same procedure applies as described at the RTT-W message. No information is lost.

TMDTA1 Too many cross section data: the memory of the computer is too small for this case.

TMEXL 1 Too many excited levels.

TMTME 1 Too many transfere matrix elements.

These two messages are both due to undetected error in the cross section set or machine errors when writing or reading tape B5. The best is first to run the same case again without making any change.

TM = -T1 Cross section error, the transfer matrix transfers into non-existent groups.

WR.C01 Wrong comment: the first 10 words of the second jobt title card do not coincide with the first job title card (see Sect. 10.2., description of the data cards).

WR-GM The geometry programme combined with the RWS programme is not contained in the geometry list. This may only happen if one combines a new geometry routine with the RWS programme. The name of this new geometry routine has to be listed near the storage location QRCS in the RWS programme.

WR.QC Attention! In the specification list a specification is detected which does not contain the final *. The specification is <u>not</u> omitted. Exception: if the final character is a blank, this signals the end of the specification list. All further information is omitted.

Also the geometry programme can produce error messages when testing the geometry input data. The messages are described in the geometry part of the manual.

Furthermore it may be possible that the programme stops at HTR instructions. Such steps are programmed to avoid that the square root of a negative number is taken. Either undetected errors in the input data may cause this, or, if it appears in the final print out, it is usually due to rounding errors when calculating the variance of several results. But it happens only if one makes a run with a very low history number. A stop during reading the input data at a location $< 3000_8$ (under the assumption, that the monitor loads at 144_8) results from an inproper format of the data cards.

10.6 The History Dump

By using the DUMP * option or by setting the Sense Switch 4 down on the operators console one can get an internal history dump. Care must be taken, because the use of this possibility usually produces an enormous amount of output. It should therefore only be used when testing new features of the programme, because one single neutron may produce up to four or five pages of output. The appearing labels have the following meaning:

STARTING POINT H = h. It signals that a primary neutron history is started where H indicates the current history number. If the ITER * option is used, it may be followed by: PRIMARY NEUTRON HISTORY. In this case the contribution of the history is not counted in the final result.

COLLISION PROCESS and BOUNDARY CROSSING are self explanatory.

If after a collision process the neutron history continues (and is not terminated by a Russian Roulette procedure for example), the type of process and the corresponding isotope number K are printed, i.e.

ELAST. COLLISION, ISOTROPIC K = k
ELAST. COLLISION, ANISOTROPIC K = k
INELAST. COLLISION, EXCITED LEVEL K = k
INELASTIC COLL., TRANSFER MATRIX K = k
INELASTIC COLL., EVAPORATION MOD. K = k

They are followed by detailed information on the different parameters of the neutron history. These consist of: RG = l the region number, and if the SORSEC geometry is used, three further quantities are of interest:

RGH , ARG and SRG (explained in the Geometry Manual), further I = i index of the energy group, E = E energy of the neutron after the collision, W = W neutron weight factor, PTX = D track length in cm from the previous event to the boundary crossing point; in the case of a collision it is zero, PTD = D track length in cm from the previous event to the collision point; in the case of a boundary crossing it is zero. Note, if the last event terminated the history, the travel distance of the neutron appears in the starting parameter of the following neutron history. MFP = λ is the mean free path or its remaining portion, if boundary crossings were encountered. SGT = Σ_{TO} is the total macroscopic cross section of the region under consideration (after a boundary it is the cross section of the new region). C.T = $\cos\Theta$, S.T. = $\sin\Theta$ and C.P. = $\cos\phi$, S.P. = $\sin\phi$ are the azimuth and horizontal cosine and sine values of the flight vector. Z = Z, Y = Y and X = X are the space coordinates of a boundary crossing or a collision. T = t actual time an event occurs, TZ = t_o starting time of the history under consideration.

If the collision takes place in a zone containing fissionable materials the following quantities regarding the production of fractional neutrons are printed: PSI(K) = $\psi(k)$ total fractional generated weight mod (1) or mod (k_{eff}^{flux}) if ITER * was specified of isotope K (=k), KEFF = k_{eff}^{coll} total (unnormalized) generated weight (including the fractional weight produced at this fission. T = t time of the weight production, TZ = t_o starting time of the history under consideration.

If PSI(K) exceeds 1.0 (or k_{eff}^{flux}) it is reduced by 1.0 (or k_{eff}^{flux}) and a line with STORE SEC. NEUTRON appears, where K = k isotope number, X , Y, Z are the coordinates of this event and WFSUM(K) = $\sum_j \psi_j(k)$ the total weight

of fractional neutrons produced for isotope K. If the ITER * option is used WFSUM(K) is usually not an integer.

BUFFER STOR. LOC. = (I4) gives the position of the stored neutron parameters in the buffer storage.

It runs cyclically from 1, 2, 3 ... to 401, 1, 2, etc.

A history can be terminated by the following reasons:

a) NEUTRON HISTORY FINISHED BY LOW ENERGY LI.

b) NEUTRON HISTORY FINISHED BY RUSS. ROULETTE OR TOO SMALL WEIGHT.

c) NEUTRON HISTORY FINISHED BY LEAKAGE, if the neutron enters a region specified by the LEAK * option. LWS = $\sum_{h} W_{hj}^{LE}$ sum of the total weight lost by leakage including this last event.

d) NEUTRON HISTORY FINISHED BY TOO LONG TIME, if the time parameter exceeds the value specified by the TMAX * option. TMX = t_{max} upper time limit to which a history is followed.

After the termination of a history has been signaled by one of the above mentioned messages the following parameters are printed (they were already defined earlier) : RG, RGH, ARG, SRG, I, E, W, Z, Y, X, T, TZ.

Now a new history starts. It can either be a primary neutron as described above, or a secondary neutron which is labeled: STARTING POINT FISSION NEUTRON with the parameters K = k isotope index, HS = n index number of the secondary neutron history.

If specifications are used, the following additional quantities are dumped:

a) RURU ★ (Russian Roulette) option:

NEW WEIGHT BY RUSSION ROULETTE, appears if after a Russian Roulette procedure the weight factor is increased (usually to unity).

W: new weight factor

OLD W: old weight factor

b) AGE ★ option (for calculating the slowing down length)

In addition to the parameters printed at each collision point the following quantities appear:

ZAG = x_o, YAG = y_o, XAG = z_o: they give the coordinates [cm] of the collision point relative to the origin of the neutron history under consideration. These coordinates present always the true distance to the starting point, while the coordinates X, Y, Z in infinite lattice geometries refer to the projection of the position into the unit cell.

c) ELP ★ option: the ""expected Leakage Probability" version of TIMOC:

If this option is specified, first the flight vector is followed from the starting or a collision point to the outer boundary. Such a boundary crossing is labeled: BOUNDARY CROSSING, SPEC. ELP ★. In the following parameter print out PTX is the distance from the <u>last</u> event to the boundary crossing under consideration. If the outer or leakage boundary is reached BOUNDARY CROSSING IS TO LEAKAGE is signaled. The distance to this last boundary crossing is printed together with the parameters of the next collision process.

WEIGHT BEFORE COLL. = W_{old} is printed before a collision process in the ELP ★ version. LEK = $W_{h,j}^{ELP}$ weight fraction lost by leakage at the last collision process. LWS = total weight lost by leakage.

The following parameters have in the ELP * version in the case of a COLLISION PROCESS a different meaning: PTX distance from an event to the leakage boundary of the foregoing flight vector calculation, PTD distance from the last collision process to this one. MFP in the case of calculating the length of the flight vector from a collision process to the outer (leakage) boundary this quantity is set to 10^{20}. In the case of a collision process it contains the distance in mean free paths from the previous collision point to the one under consideration.

d) SMEC * option: TIMOC version to calculate small effects. If this option is specified PSI refers only to the multiplications caused by the so-called unperturbed neutron histories. KEFF is however the normalized multiplication factor counting all neutrons generated in the system.

10.7 The Re-Run Procedure

During programme execution every 1000 histories (primary histories in the Standard Version, secondaries if ITER * is specified) a re-run (or continuation) dump is written on tape B5. It permits a continuation of the problem using all accumulated data. This feature allows an uncompleted problem (uncompleted because of machine malfunction or a running time limit) to be completed at a later time. It also allows a completed problem to be extended to a larger number of histories.

For a continuation the dump tape is mounted on B5. There is no link 1 (Nuclear Data Input) and only a very limited link 2 (Random Walk Sampling

Programme) input. The Re-Run Programme consists of a Loader and the following sequence of cards:

CARD	COLUMNS	FORMAT	SYMBOL
1	1 - 6	I6	NF NF: LOADER reads file NF which corresponds to link 2 (Random Walk Sampling Programme) from System Tape A5. $0 < NF \leq 9$. NF must correspond to the file number of the original RWS programme of which the re-run is made.
	1 -60	10A6	TITLE TITLE: Title card to identify the job. It must agree with the Title Card of the original job of which the re-run is made.
	61-72	I12	HMAX or FISMX (if ITER * is specified) HMAX (FISMX): Cumulative number of histories to be calculated (includes the number of histories already calculated in the original RWS computer run).
3	1 - 6	I6	NF $1 \leq NF \leq 9$: LOADER reads file NF which corresponds to link 1 or link 2 problem. NF = 0 or blank: the execution is terminated and all tapes in use are unload.

11. THE "TIME TAPE SCANNING" PROGRAM

11.1 Optional Modes and Specifications

As already mentioned in chapter 7, the purpose of the TIME TAPE SCANNING (TTS) program is to analyse the neutron histories with respect to the time variable. The corresponding data are read from tape B6 into the computer and are stored according to the input specifications. The TTS program may also be used to calculate angular dependent fluxes as well as some time-independent quantities which cannot be obtained from the RWS program (collision density etc.). The different possibilities of the TTS program are controlled by options. We list them in alphabetical order, together with a short description.

CODN * The time dependent collision density and flux (calculated from the data of the collision density by dividing each neutron weight by the total macroscopic cross section) is printed. The collision events must be recorded on the Time Tape.

COMM * Allows a change of instructions in the program during data input. Exactly the same procedure as in the RWS program. Card(s) 8 are required.

DPAN * This option allows the calculation of the angular dependent transmission, current or flux at the boundaries between different regions. The range of $\cos\Theta$, where Θ is the angle between the direction of flight and the surface normal, is divided into 20 equal intervals. This option may only be used in connection with the FLUX * or NONE * option. The boundary crossings must be recorded on the Time Tape.

DUMP * Gives a dump of all events which are recorded on the Time Tape. The same is achieved by putting Sense Switch 4 down.

ENGB * Gives the opportunity to insert new energy boundaries, otherwise those defined together with the cross sections are used. Requires card 4.

FLUX * Calculates the time dependent transmission and the flux (neutron weight divided by cos Θ , where Θ is the angle between the direction of flight and the surface normal) in both directions on the boundaries between different regions. The boundary crossings must be recorded on the Time Tape.

FXTM * Calculates at fixed time points the number of neutrons present in each region. Multiplication of each neutron weight with the velocity of the neutron gives the flux. Boundary crossings and collisions must be recorded on the Time Tape.

GRVE * The opportunity to insert new group velocities in connection with the ENGB * option. Only useful if the option FXTM * is used. Requires card 5.

NOCM * The angular dependence of the information contained on the Time Tape is suppressed.

NONE * Signifies that no other specification is used. Time-dependent transmission and current (neutron weight multiplied by cos Θ) on the boundaries between different regions are printed. The boundary crossings must be recorded on the Time Tape. This specification was introduced because the TTS program expects at least one specification.

REGN * The output is only produced for those regions, which are explicitly listed in the input. Card 6 is required.

TFLX * Calculates on the basis of the transmission at each boundary crossing the time and energy integrated flux as well as the first time moment of these quantities. The boundary crossings must be recorded on the Time Tape.

UNEN * Calculates the energy spectrum of the transmission, current and flux at the boundaries. To be used in connection with the option FLUX *.

11.2. Hierarchy of the Options

Since several options interfere with each other, a certain hierarchy of priority has been established. In the following, the main options are listed in this hierarchical order together with those options which are automatically suppressed. Note that the input data, which are required by an ignored option, must nevertheless be present, because the hierarchy is checked after the whole input is made.

1.) NONE * All other specifications are ignored.

2.) FXTM * The specifications CODN *, FLUX * and TFLX * are ignored.

3.) CODN * The specifications FLUX * and TFLX * are ignored.

4.) FLUX * The specification TFLX * is ignored.

5.) TFLX *

6.) NOCM * The specification DPAN * is ignored.

11.3 Input Data and Formats

CARD	COLUMNS	FORMAT	SYMBOL
1	1 - 6	I6	NF NF: LOADER reads file NF which corresponds to Link 3 (= TTS = TIME TAPE SCANNING program) from the Systems Tape A5, $0 \leq NF \leq 9$. If NF = 0, the executions is terminated and all tapes are unloaded.
2	1 -60	10A6	TITLE TITLE: Title card to identify the job. It must agree with card 2 of the NDP and card 2 of the RWS program. Column 72 must contain the logical file number of the data, which are stored on the Time Tape B6. This number appears in the print out of the title card of the corresponding RWS program. See chapter 11.4.

All the following cards are read by the WPK06 input routine (for description see Ref. (27)) and must therefore be written in the corresponding formats.

3	8 -10	BCD	Columns 8-10 contain the symbol BCD.
	12	i	N N: Number of 6-character words following in columns 13 - 72.

CARD	COLUMN	CONTENTS	SYMBOL	
	13-72	n(a6)	All specifications and otpions which are used in this run (see chapter 11.1).	
3a	8-10	TRA		
	12-14	4,4		
4	8-10	DEC	Only if ENGB * is specified.	
	12 cont.	i	N	
			N: Number of energy groups. $0 < N \leq 50$.	
		f	E(N)	Boundaries of energy intervals.
		f	E(N-1)	They have to be in decreasing order
		.		
		.		of magnitude
		f	E(O)	
4a	8-10	TRA		
	12-14	4,4		
5	8-10	DEC		Only if GRVE *'is specified.
	12 cont	i	N	Number of group velocity parameters. Must be equal to the number of energy groups.
		f	T(N)	Average neutron velocities.
		f	T(N-1)	They have to be in decreasing order
		.		
		.		of magnitude.
		.		
		f	T(1)	
5a	8-10	TRA		
	12-14	4,4		

CARD	COLUMN	CONTENTS	SYMBOL	
6	8-10	DEC		Only if REGN * is specified.
	12 cont	i	R(1)	Different region numbers in arbitrary
		i	R(2)	order. Note that only for those regions,
		.	.	
		. .	.	which are listed here, the output is
		.	.	
		.	.	printed.
6a	8-10	TRA		
	12-14	4,4		
7	8-10	DEC		
	12 cont	i	N	
			N: Number of intervals of the time distri-	
			bution. For the FXTM * option, number of	
			time points. $0 < N \leq 50$.	
		f	T(N)	Boundaries of the time intervals.
		f	T(N-1)	For the FXTM * option, fixed time
		.	.	
		.	.	points (T(0) must be omitted).
		.	.	
		f	T(1)	The quantities have to be in de-
		f	T(0)	creasing order of magnitude.
7a	8-10	TRA		
	12-14	4,4		
8	8-10	OCT		Only if COMM * is specified. There must be
				as many pairs of cards 8 and 8a (8b) as
				there are instructions to be changed.

CARD	COLUMNS	CONTENTS	SYMBOL
	12 cont	i	I I is the octal address of the instruction to be replaced. It is the address in the listing and is relocated in the computer.
		i	J J: Octal representation of the 36 bit string replacing the contents of address I in the original version of TIMOC. If the new instruction contains an address, the absolute machine address has to be used. Since in Fortran II, Version 2 the TTA program is loaded at location 3000 (octal), one has to add this number to all addresses.
8a	8-10	TRA	If the foregoing card is not the last change,
	12-14	4,4	else card 8b.
8b	8-10	TRA	
	12-14	3,4	
9	1- 6	I6	NF NF: Loader reads file NF which corresponds to a Link 1, 2 or 3. $0 \leq NF \leq 9$. If NF = 0 or blank, the execution is terminated and all tapes are unloaded.

11.4 Some operational features

A Time Tape can contain the data of several RWS runs. Besides the title card, which may be equal for these runs, the respective data are distinguished by a logical file number, which is set by the RWS routine and must be included in column 72 of the title card of the TTS program (card 2 of the input data). Under the control of the LOADER routine, the first RWS run (Link 2) rewinds the Time Tape B6 and assigns the logical file number 1 to the data to be written on the Time Tape. At the end of each RWS and TTS run (Link 3), the Time Tape remains in its position, so that a following RWS run is able to put new data in the correct position and to increase the logical file number by one. Thus, for example, if the Time Tape contains the data of 4 RWS runs, a new data file may be included by spacing the Time Tape in the correct position using a TTS run of logical file number 4. The maximum number of logical files on the Time Tape is 9.

If the Time Tape is full, it is possible to continue the run with another tape. The message END OF TAPE, STOP 7777 PLEASE NEW TAPE ON B6 AND DEPRESS START KEY appears on and off line. In the TTS program, a similar message with ..... STOP 66666 PUT OTHER TAPE ... is printed on and off line if an end-of-tape is sensed during the scanning. It is not recommended to use this feature because experience has shown that it is very easy to mix up the different tapes, with the result that no valuable output is produced. It is better to divide the job into several smaller ones, where the data of each fit on one Time Tape and to combine the results by hand.

It is simple to estimate that proportion of a Time Tape, which is occupied by data. At the end of each RWS run - if the option TIME * was used - the

label TOTAL NUMBER OF RECORDS ON TIME TAPE = (I6) appears. The length of a full tape is of the order of 4000 or more records. Such a record consists of the data of 100 events (boundary crossings, collisions or starting neutrons), and each event occupies 5 words (time, energy, weight and several other parameters which are fed in the remaining 2 words). The length of such a record is thus 500 words.

11.5 The Initial Output

The first line of each page of the output of the TTS program contains the name of the code, TIMOC, and the job title as specified on the title card (see card 2). Further, the following is printed.

1.) All specifications (see ch. 11.1)

2.) Repetition of the geometry data. This print out is exactly the same as in the RWS program.

3.) Only if ENGB * is specified. The label ENERGY BOUNDARIES, IM = , followed by the number of energy groups and the corresponding boundaries, appears as specified in the input data.

4.) Only if GRVE * is specified. The label GROUP VELOCITIES, IM = , followed by the number of energy groups and the corresponding group velocities, is printed as specified in the input data.

5.) The next print out is TIME BOUNDARIES, TM = , followed by the number of time intervals (or time points, if the option FXTM * is used) and the corresponding interval boundaries (time points), is printed as specified in the input data.

6.) Only if DUMP * is specified or the sense switch 4 on the operator's console of the computer is "down". Care must be taken, because the use of this option produces normally an enormous amount of output, since the whole contents of tape B6 is printed. For each record on the tape B6 (which contains 100 events, each event being described by 5 words), the following title appears: TIME ENERGY WEIGHT PROJ(W/SGT) SIGN CM(ANG) CM(SRG) 4 2 1 RG TM IM INDEX N. The first three columns contain obviously the time, the energy and the weight of the neutron at the event under consideration. As event, we consider the following possibilities: (i) boundary crossing, (ii) collision and (iii) start of a history. In the case of a collision, the weight of the neutron is that before the collision. The column PROJ(W/SGT) contains in the case of a boundary crossing $\cos\Theta$, Θ being the angle between the direction of flight of the neutron and the surface normal. In the case of a collision, the quantity "weight over total macroscopic cross section" is printed. This may (optionally) be used for the determination of the collision density. The SIGN column contains either 0 or -0. 0 means that the present event is a boundary crossing, while -0 means collision. The contents of the column CM(ANG) are only different from zero if there is a boundary crossing, if in the RWS programme (link 2) the DPAN * option was taken and if in the TTS routine none of the options FXTM *, NOCM * or CODN * are used. The integer appearing there corresponds to the angle interval of the direction of flight of the crossing neutron. It is calculated by taking the integral part of 10 times the contents of the column PROJ(W/SGT). The contents of the columns CM(SRG) is only different from zero if in the corresponding RWS program the SRRG * option was used in

connection with the SORSEC geometry routine. In the case of a collision this column contains the sector number SRG (defined in the description of the SORSEC geometry routine) where the collision takes place. In the case of a boundary crossing through the window of the SORSEC geometry, it contains for outgoing neutrons the previous sector number; for incoming neutrons the new one. In the case of other boundary crossings, the previous sector number is printed. Sector numbers $\geq$ 10 are put to zero for convenience. The next three quantities are labelled with 4, 2 and 1. This means sybolically the position in the word where they are incorporated during the transmission of the data, namely in the tag bits 4, 2 and 1. These quantities consist therefore of only 0 and 1 and are usually zero. If under tag 4 a 1 appears, then all quantities in this line are only of importance in connection with the FXTM * option. This can happen only if there is a boundary crossing which is not counted under the TRAN * option of the RWS program, for example, a crossing through surfaces other than the window in the SORSEC geometry. A 1 under the tag 2 column signifies that this line contains information about the angular distribution of a neutron when crossing through a surface and is therefore only of interest in connection with the DPAN * option. The tag 1 column contains a 1 if the data in this line refer to the start of a new history or to a collision in the ELP * option of the RWS program. This is only of interest when the FXTM option is used. The RG column contains in the case of a collision the region number where the collision takes place. In the case of a boundary crossing it contains either the PRG (previous region) or the RG (region) number, depending on the sign of PROJ (PROJ $>$ 0 $\rightarrow$ PRG, PROJ $<$ 0 $\rightarrow$ RG). If the FXTM * option is used, this

column always contains the PRG number. The columns labelled with TM and IM contain the time and energy indices of the present event. They are determined from the corresponding boundaries of the time and energy intervals. Note that the indices increase with increasing time and energy. The column with INDEX contains a number specifying the position of the data in the matrix, which is set up for the final output. This number is determined from M*((((TM-1)*(LMA+1)+LM)*(CMA+1)+CM)*IMA+IM) where TM is the time index corresponding to the actual time, LMA+1 is the maximum number of the geometrical regions in this sample and LM is the actual region index (which is labelled with RG), CMA+1 is the maximum number of angular intervals or sector regions and CM is the corresponding actual index, IMA is the total number of energy groups and IM is the actual energy index, M=1 for the FXTM * option, M=2 for the CODN * or the TFLX * option and M=4 for the other options. The last column with the heading N signifies the actual number of contributions of the present data to the data matrix and contains the integers 0 or 1 except for the FXTM * option. There it may be different from 1 since between several events there may ly none or several time points of interest. The number in the columns labelled with TM and INDEX are in this case those corresponding to the last contribution.

7.) NUMB. OF PRIM. HISTORIES = I7, NORMALIZATION = F11.3, GEN.SEC. = I7(I7), CALC.SEC. = I7 (I7). These quantities mean (i) the number of primary neutrons handled, (ii) the total weight, which is used for the normalization of all quantities to one source neutron (or, if in the RWS program the ITER * option was used, to one secondary neutron), (iii) the number of generated secondary neutrons and (iv) the number of secondary

neutrons whose histories were actually followed up to the moment of truncation. For more extensive explanations, see ch. 10.4.2 for (i), (iii) and (iv) and ch. 10.4.1 for (ii).

8.) I10 CORRECT RECORDS, I8 WRONG RECORDS. This label signifies the length of the data on tape B6. One record contains the information of 100 events, each event being described by the contents of 5 words. Thus one record has the length of 500 words. On a normal tape there is place for about 4000 to 5000 words of this type. The number of wrong records refers to a redundant part on the tape. If in tape-writing a redundancy is encountered, this part is overwritten by short records of one word length until the wrong part is over. In reading, these short records are counted in the above label.

11.6 Error Messages

The program performs a number of checks and tests during the data input and the calculations. They control proper data input and tape handling (data transmission).

The error message is composed of the label TYPE OF ERROR (A6), LOCATION (O6) **** followed either by CALC.CONTIN. (calculation continued) or CALL EXIT. The A6-word signalizes the kind of error. The meaning of this 6-character string is given in the list below. The O6-location contains an octal integer refering to the storage location of the error stop in the listing of the TTA program.

The different 6-character strings are listed in alphabetical order.

1LOST The information contained in a 500-word record is lost because of a redundancy on tape B6.

3LOST1 The information contained in three 500-word records is lost because of redundancies on tape B6. The calculation is terminated.

EN.DT1 The numbers given as new boundaries of the energy groups are not in the right order. See description of input card 4.

EOF-N1 Instead of input data, an end-of-file is sensed on the monitor input tape A2.

GDTRT1 A redundancy was encountered in reading the geometrical and energetical specifications from tape B6.

GV-DT The input data for the group velocities are not in the right order. See description of input card 5. Those group velocities, which are transmitted from the TWS program are taken for the present calculation.

NBHEF Instead of the number of histories, an end-of-file is sensed on tape B6. The number of histories is put to unity for this run and the calculation is continued.

NBHST Instead of the number of histories, a redundancy is sensed on tape B6. The number of histories is put to unity for this run and the calculation is continued.

NO PLC The data matrix for the output is too large to fit in the computer.

NODAT1 Instead of the geometrical and energetical specifications, an end-of-file is sensed on tape B6.

NOFIL1 The program finds no end-of-file (which signalizes the termination of the data) on tape B6.

NOFRT1 The program cannot find the desired data on tape B6. Probably a wrong tape was put on the tape unit.

RTT-N1 A redundancy is encountered on the monitor input tape A2.

TI.DT1 The input data for the time boundaries (points) are not in the right order. See description of input card 7.

11.7 The Final Output of Sample Values

11.7.1 General Remarks

The TTS program can produce five groups of time dependent output quantities, which are obtained by the use of certain options.

In the following, for a time dependent quantity, as much as possible reference is made to the description of the corresponding stationary quantity. For example, the sampling formula will be omitted here if it differs from the corresponding stationary one only by the appearance of the time variable and a division over the specified time interval $\Delta t_g = t_{g+1} - t_g$.

In four of these five groups, the output is obtained by collecting the data in the preassigned time intervals. In these cases, the label TIME INTERVAL = (E13.5) - (E13.5) together with the boundaries is printed. The fifth group specifies data at fixed given, time points. Then, the label TIME POINT = (E13.5) appears on the output.

The normalization is in all cases made by the quantity NORMALIZATION, which is described in chap. 11.5. For more detailed information about this quantity, which will here be denoted by S, see chap. 10.4.1. Therefore, in the following, no distinction will be made between the Standard Version, the ITER * Version and the ELP * Option of the corresponding RWS program. Also the summation over the histories will be described only by a single sum. Its meaning is, according to chap. 10.4.1, the sum over all secondaries in the ITER * Version and over all primaries and secondaries in the Standard Version of the corresponding RWS program.

11.7.2 Transmission and Current

These quantities are exactly the same as those described in chap. 10.4.17 except for the explicit appearance of the time variable. In the corresponding RWS program, the TRAN * option must be used and the boundary crossings must be recorded on the Time Tape B6. Here, the transmission and the current are obtained at the boundaries between two regions using the NONE * option. Note, that this option must be omitted if there is any other option!

These quantities are labelled by DISTANCE CI ENERGY GROUP TRA(IN/OUT) TRA(OUT/IN) TRA(SM) CUR(IN/OUT) CUR(OUT/IN) CUR(SM). Only those quantities

which differ from those in chap. 10.4.17 are described here. The column labelled with DISTANCE contains in the case of the GSPHER geometry the actual radius of the sphere which is crossed by the neutrons. In the case of the SLBCYL geometry, it contains the radius of the circle in the cylindrical case and the distance from the origin in the slab case. In the case of the SORSEC geometry, it contains (as floating point number) the region number in the window where the crossing takes place. The column labelled with CI is used to obtain angular dependent output. It is necessary to specify in both the RWS and the TTS program the DPAN * option.

For convenience, we give here the explicit sampling formula for $Q_{CUR+}(l,i,g)$ $\left[= \text{CUR(IN/OUT)} \right]$ in the time interval $\left[t_g, t_{g+1} \right]$:

$$Q_{CUR+}(l,i,g) = \left[\sum_h \sum_j W^+_{h,j}(l,i,g) \cos \Theta_l \right] \cdot (S \cdot F_l \cdot \Delta t_g)^{-1} . \qquad (11.1)$$

The other quantities are similarly obtained, see chap. 10.4.17.

If the UNEN * option is used, a second table with the label PER UNIT ENERGY is printed, where the above quantities are divided over the respective energy interval $\Delta E_i = E_{i+1} - E_i$. Thus, for example,

$$\text{TRA(IN/OUT) (PER UNIT ENERGY)} = \frac{Q_{TRA+}(l,i,g)}{\Delta E_i} . \qquad (11.2)$$

11.7.3 Transmission and Flux

Using the FLUX * option, one gets the flux instead of the current. It is obtained by sampling the quantity "weight over $\cos\Theta_l$ " at each boundary crossing. The following label appears: DISTANCE CI ENERGY GROUP TRA(IN/OUT) TRA(OUT/IN) TRA(SM) FLX(IN/OUT) FLX(OUT/IN) FLX(SM).

The first six labels contain the same quantities as in the previous chapter. The quantity $Q_{FLX+}(l,i,g)$ $\left[= \text{FLX(IN/OUT)}\right]$ is sampled from

$$Q_{FLX+}(l,i,g) = \left[\sum_h \sum_j \frac{W^+_{h,j}(l,i,g)}{\cos\Theta_l}\right] \cdot \left(S \cdot F_i \cdot \Delta t_g\right)^{-1} . \qquad (11.3)$$

The quantity under the label FLX(OUT/IN) is obtained by replacing in (11.3) $W^+_{h,j}(l,i,g)$ by $W^-_{h,j}(l,i,g)$ and

$$\text{FLX(SM)} = \text{FLX(IN/OUT)} + \left|\text{FLX(OUT/IN)}\right| .$$

For the use of the UNEN * option, we refer to the previous chapter.

11.7.4 Time Integrated Flux

The Time Integrated Flux can be obtained from the boundary crossings using the TFLX * option. It allows one to sample

$$Q_{TFL}(l,i,g) = \int_{t_g}^{t_{g+1}} t\,\phi(t,E)\,dt \qquad (11.4)$$

and

$$Q_{FL}(l,i,g) = \int_{t_g}^{t_{g+1}} \phi(t,E)\,dt \qquad (11.5)$$

which are the first moment and the mean of the flux with respect to the time variable.

In the printout, the label DISTANCE CI ENERGY GROUP F*T*DT*DE F*T*DT F*DT*DE F*DT F*T*DT/F*DT appears. The first three labels are the same as in chap. 11.7.2. The column labelled with F*T*DT contains just (11.4) and is sampled from

$$Q_{TFL}(l,i,g) = \left[\sum_h \sum_j \frac{t_{h,j} W_{h,j}(l,i,g)}{\cos\Theta_l}\right] \cdot (S \cdot F_l \cdot \Delta E_i)^{-1} \qquad (11.6)$$

where $\Delta E_i = E_{i+1} - E_i$. The quantity labelled with F*T*DT*DE is the energy integral of (11.6), $Q_{TFL}(l,i,g) \cdot \Delta E_i$. Similarly, the column labelled with F*DT contains (11.5) and is sampled from

$$Q_{FL}(l,i,g)=\left[\sum_{h}\sum_{j}\frac{W_{h,j}(l,i,g)}{\cos\Theta_l}\right]\cdot\left(S\cdot F_l\cdot\Delta E_i\right)^{-1}. \qquad (11.7)$$

The column with F*DT*DE contains $Q_{FL}(l,i,g)\cdot\Delta E_i$ and the last column, which has the label F*T*DT/F*DT, contains the ratio $Q_{TFL}(l,i,g)/Q_{FL}(l,i,g)$. This quantity is very sensitive to whether or not the neutron population has reached the equilibrium state. If the flux decays exponentially,

$$\phi(t,E) = e^{-\alpha t}\,\psi(E)$$

and we take in (11.4) and (11.5) $t_g = t_o$ and $t_{g+1} = \infty$, then

$$\text{F*T*DT/F*DT} = \frac{1}{\alpha} + t_o$$

is <u>independent</u> of the energy.

11.7.5 Collision Density and Flux

If the CODN * option is used, we obtain the collision density and the flux based on this quantity. The collisions must be recorded on the Time Tape B6.

The following label is printed: RG CI ENERGY GROUP NEUTRONS COLLDEN*DE COLLDEN CFLUX(E)*DE CFLUX(E) CFLUX(E)*E. The column

labelled with RG contains the region number, which is defined in the corresponding geometry routine. It is equal to the region number of the absorption and flux calculations in the corresponding RWS program. The column with CI contains the sector region number if in the corresponding RWS program the SORSEC geometry and the SRRG * option is used. Otherwise, this column contains 0. For a detailed description of this number, see chap. 11.5. The column with NEUTRONS contains the number ($Q_{CL}(l,i,g)$ of collisions in the region RG = l

$$Q_{CL}(l,i,g)=\left[\sum_{h}\sum_{j}W_{h,j}(l,i,g)\right]\cdot\left(S\cdot\Delta t_g\right)^{-1}. \tag{11.8}$$

The column labelled with COLLDEN*DE contains $Q_{CL}(l,i,g)\cdot V_l^{-1}$, where V_l is the volume of the region, and the column with COLLDEN contains the collision density, that is $Q_{CL}(l,i,g)\cdot(V_l\cdot\Delta E_i)^{-1}$, with $\Delta E_i = E_{i+1} - E_i$. The collision density is measured as collisions per cm^3sec and eV. The quantity labelled with CFLUX(E) is the flux calculated from the collision density and is sampled from

$$Q_{CFL}(l,i,g)=\left[\sum_{h}\sum_{j}\frac{W_{h,j}(l,i,g)}{\sum_{TO}(l,i)}\right]\cdot\left(S\cdot V_l\cdot\Delta t_g\cdot\Delta E_i\right)^{-1} \tag{11.9}$$

The quantity with the label CFLUX(E)*DE equals $Q_{CFL}(l,i,g)\cdot\Delta E_i$ and that with CFLUX(E)*E is $Q_{CFL}(l,i,g)\cdot(E_{i+1} + E_i)/2$.

11.7.6 Flux at Fixed Time Points

The last output group consists of quantities which are obtained at fixed time points rather than in a time interval. These quantities are calculated if the FXTM * option is used. The collisions, boundary crossings and the starts of new neutron histories must all be recorded on the Time Tape B6.

The following line is printed: RG CI ENERGY GROUP NEUTRONS NEUTRONS/(V*DE) FLUX*DE FLUX FLUX*E . The first three labels are explained in the previous section. The column with NEUTRONS contains the number of neutrons which are present at the time point of interest in the region RG = l . That is,

$$Q_{NEU}(l,i,g)=\left[\sum_{h}\sum_{j}W_{h,j}(l,i,g)\right]\cdot S^{-1}. \qquad (11.10)$$

The quantity with NEUTRONS/V is the number of neutrons per cm^3, $Q_{NEU}(l,i,g)\cdot V_l^{-1}$, and that with the label NEUTRONS/(V*DE) contains $Q_{NEU}(l,i,g)\cdot(V_l\cdot\Delta E_i)^{-1}$ and gives the distribution of the neutrons over the energy variable. The column labelled with FLUX contains the flux calculated on this basis:

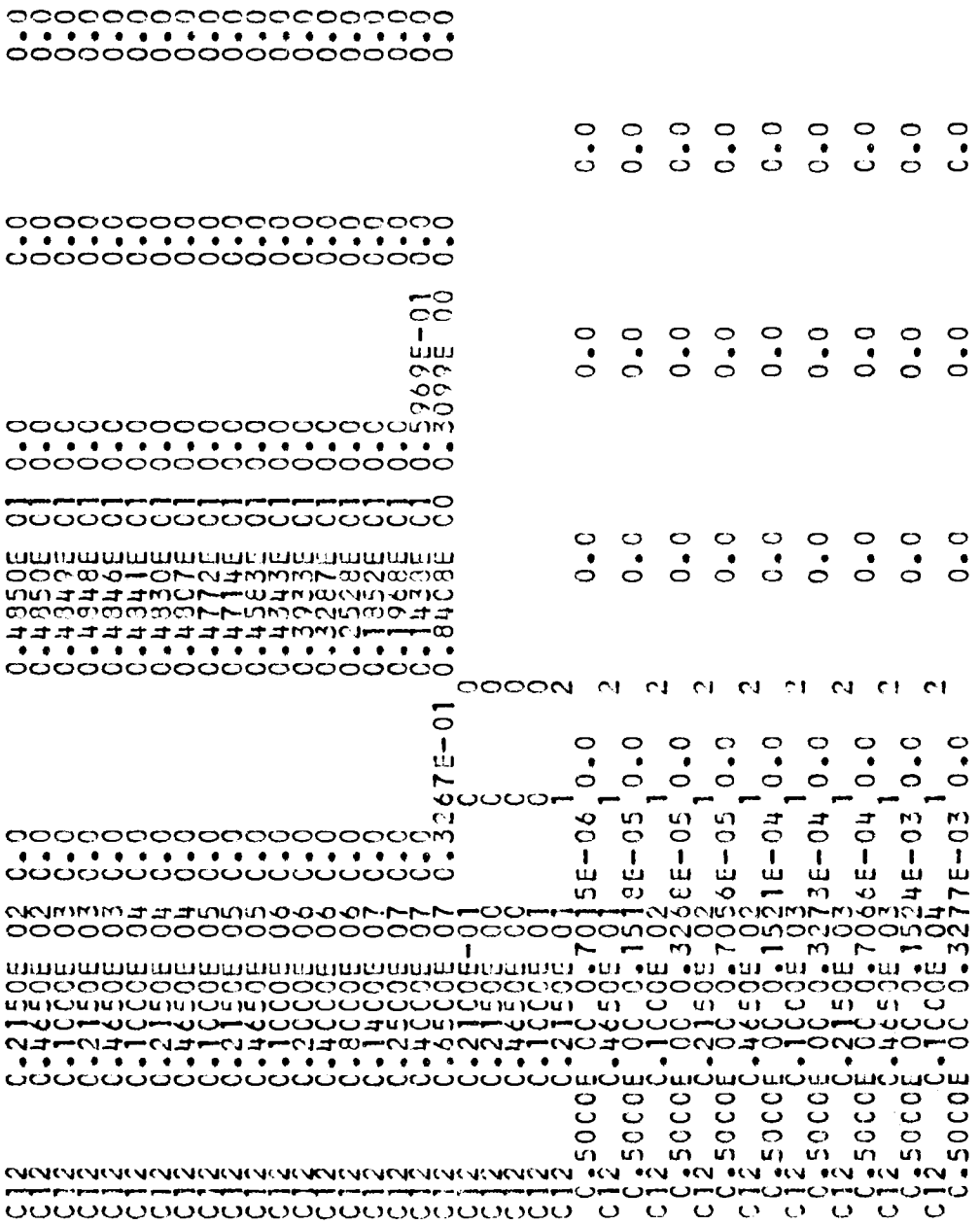
$$Q_{FL}(l,i,g)=\left[\sum_{h}\sum_{j}W_{h,j}(l,i,g)\cdot v(i)\right]\cdot(S\cdot V_l\cdot\Delta E_i)^{-1}. \qquad (11.11)$$

Here $v(i)$ is the velocity of the neutrons:

$$v(i)\ \left[\text{cm/sec}\right] = 1.3843 \cdot 10^{6} \sqrt{E_i\ \left[\text{eV}\right]}\ .$$

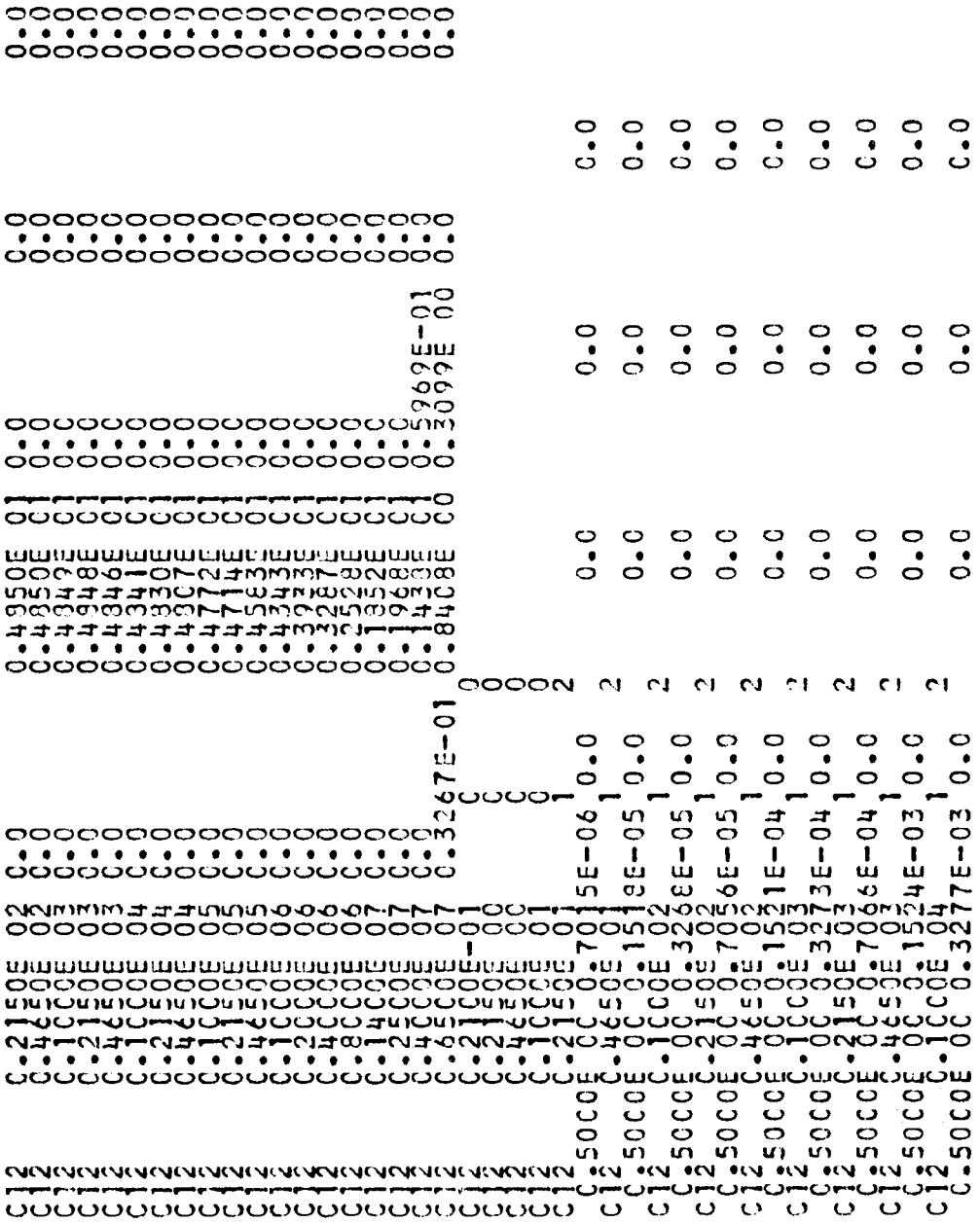
The column with FLUX*DE contains $Q_{FL}(1,i,g) \cdot \Delta E_i$ and the column labelled with FLUX*E contains $Q_{FL}(1,i,g) \cdot (E_{i+1} + E_i)/2$.

Input data of Sample Problem No. 1 (LTW-Program), continued

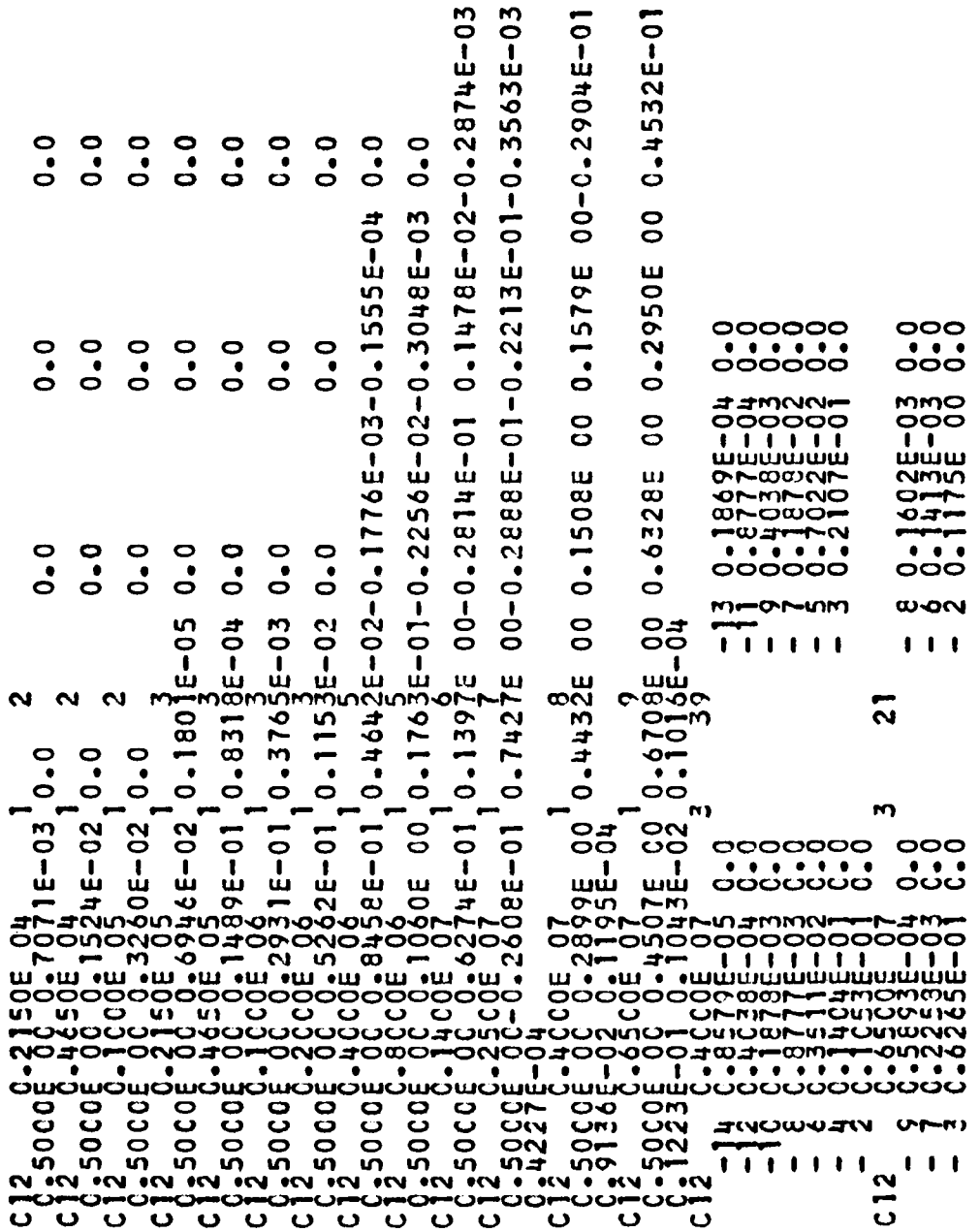


```
                5                3
               26
  0.21C0E-01
  0.2150E 00
  0.4650E 00
  0.1000E 01
  0.2150E 01
  0.4650E 01
  0.1000E 02
  0.2150E 02
  0.4650E 02
  0.1000E 03
  0.2150E 03
  0.4650E 03
  0.1000E 04
  0.2150E 04
  0.4650E 04
  0.1000E 05
  0.2150E 05
  0.4650E 05
  0.1000E 06
  0.2000E 06
  0.4000E 06
  0.8000E 06
  0.1400E 07
  0.2500E 07
  0.4000E 07
  0.6500E 07
  0.1050E 08
C12
CARBON-12 EVALUATED CROSS SECTIONS (KAPL-1965)
WEIGHTING PCIN, ACCURACY  0.100E-02, NO N-2N SCATTERING GIVEN
ELASTIC DATA GIVEN AS LEGENDRE-COEFFICIENTS, INELASTIC DATA AS MATRIX
C12      0.1191E 02          0
C12      0.21C0E-01 0.1975E-02 0.4850E 01 0.0          0.0          0.0
C12      0.2150E 00 0.9564E-03 0.4850E 01 0.0          0.0          0.0
C12      0.4650E 00 0.6513E-03 0.4850E 01 0.0          0.0          0.0
C12      0.1000E 01 0.4441E-03 0.4850E 01 0.0          0.0          0.0
C12      0.2150E 01 0.3024E-03 0.4850E 01 0.0          0.0          0.0
C12      0.4650E 01 0.2059E-03 0.4850E 01 0.0          0.0          0.0
C12      0.1000E 02 0.1405E-03 0.4850E 01 0.0          0.0          0.0
```

Input data of Sample Problem No. 1 (LTW-Program), continued



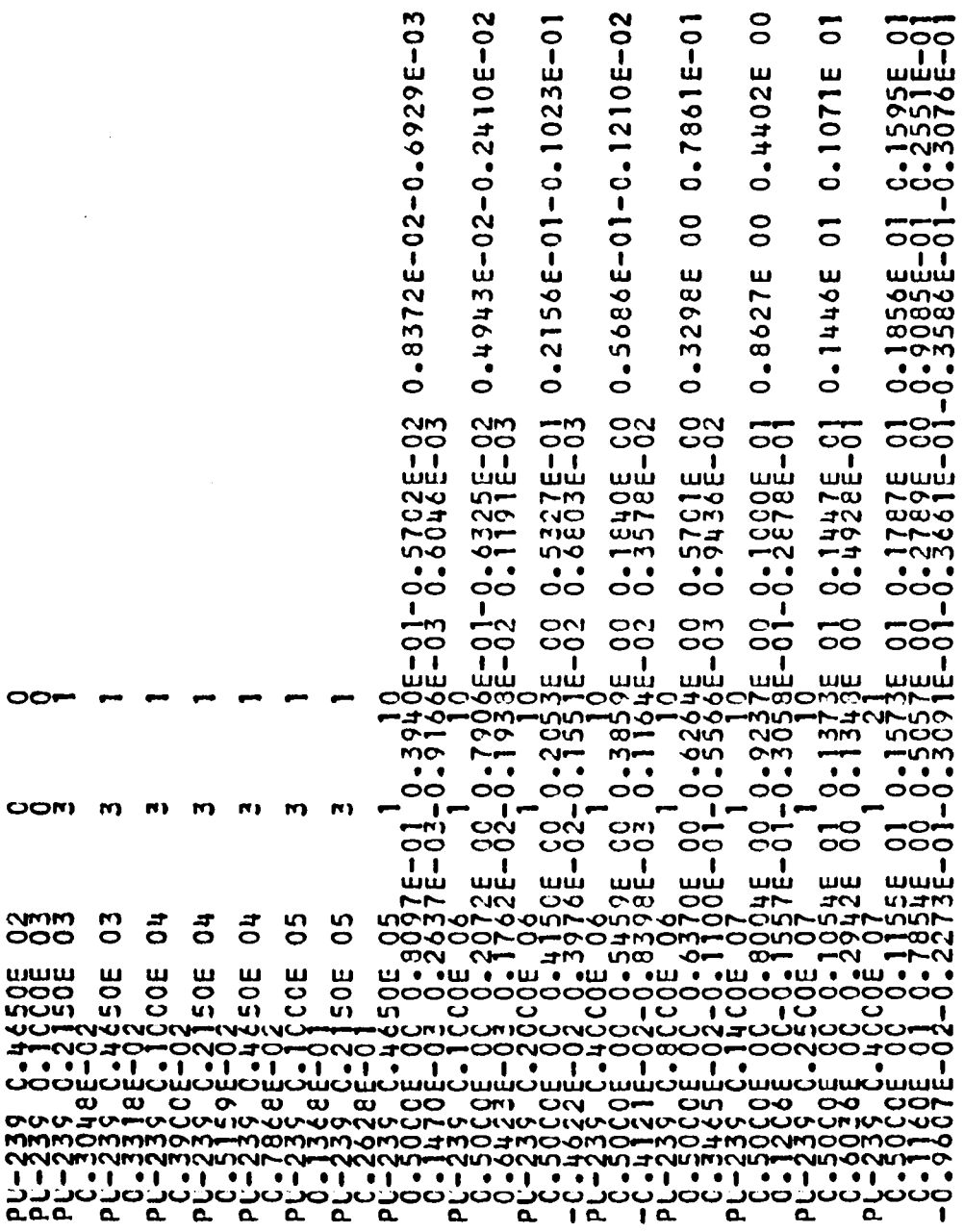
Input data of Sample Problem No. 1 (LTW-Program), continued



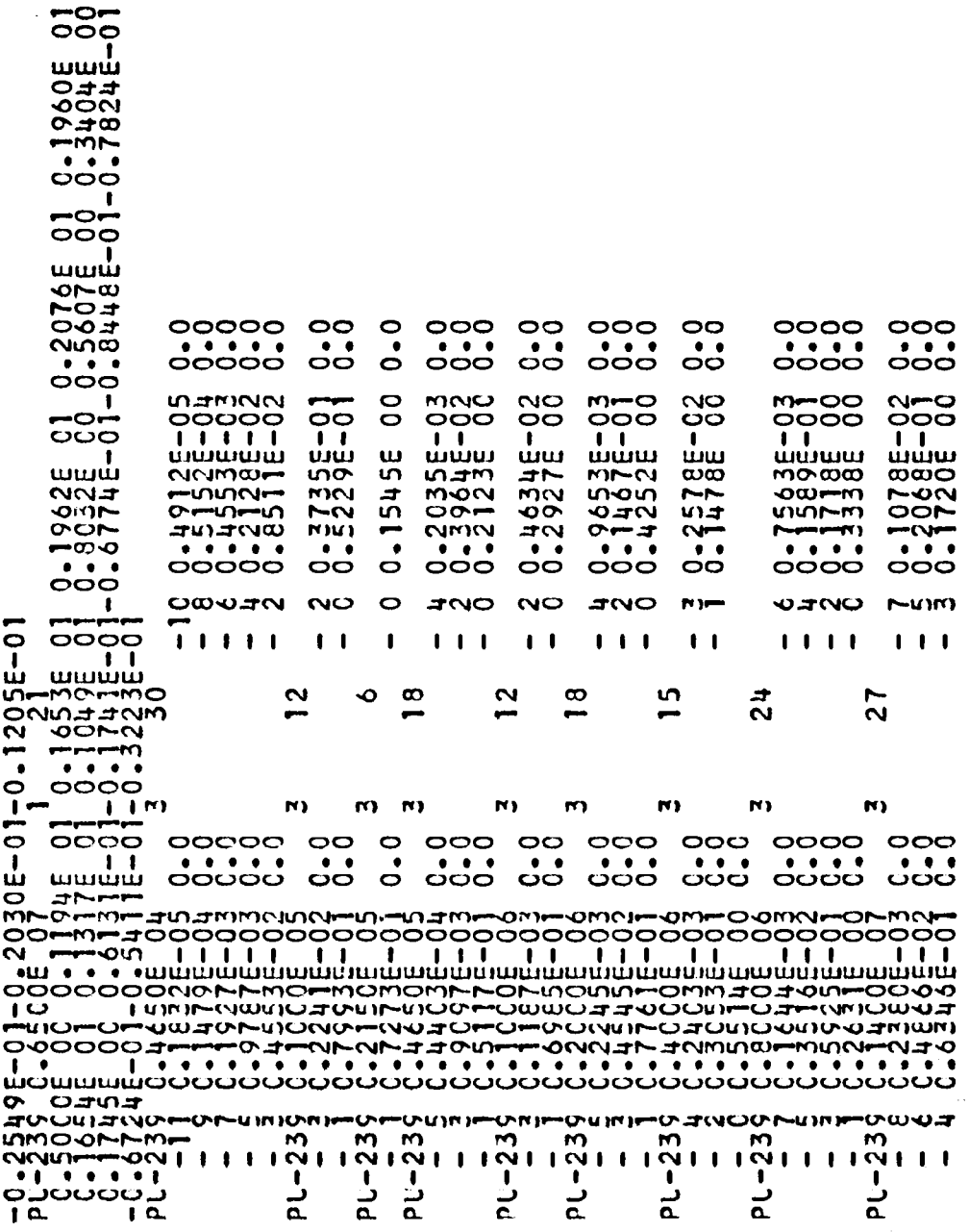
```
  - 1 C.1292E 00 C.0
PL-239
 PLUTCNIUM 239,  ENDF/B,  OCTOBER,1966.
WEIGHTING PCIN, ACCURACY  0.100E-03, N-2N SCATTERING ACCED TO INELASTIC
ELASTIC DATA GIVEN AS LEGENDRE-COEFFICIENTS, INELASTIC CATA AS MATRIX
PL-239 C.2370E 03       1
     6 C.21C0E-01 C.1050E 08 C.1410E 07 0.6739E-09 0.0
PU-239 C.21C0E-01 C.2957E 03 0.1072E C2 0.0            0.5917E 03 0.2870E 01
PL-239 C.2150E 00 C.1051E C4 0.1928E C2 0.0            0.1584E 04 0.2870E 01
PL-239 C.4650E 00 C.4080E C2 0.1345E C2 0.0            0.9243E 02 0.2870E 01
PL-239 C.1CC0E 01 C.9442E 01 0.1104E C2 0.C            0.2348E 02 0.2870E 01
PL-239 C.2150E 01 C.2C35E 01 0.10C2E C2 0.0            0.1104E 02 0.2870E 01
PL-239 C.4650E 01 C.3221E 02 0.9440E C1 0.C            0.3858E 02 0.2870E 01
PL-239 0.1CC0E 02 0.6763E 02 0.1230E C2 0.0            0.1043E 03 0.2870E 01
PL-239 C.2150E 02 C.5510E 02 0.1533E C2 0.0            0.2820E 02 0.2870E 01
PL-239 C.4650E 02 0.5360E 02 0.2450E C2 0.C            0.8157E 02 0.2870E 01
PL-239 C.1CC0E 03 C.4065E 02 C.2078E 02 0.0            0.4091E 02 0.2870E 01
PL-239 C.2150E 03 C.1763E 02 0.1884E 02 0.0            0.1988E 02 0.2870E 01
PL-239 C.4650E 03 C.2451E 01 0.1155E 02 0.0            0.5927E 01 0.2870E 01
PL-239 C.1CC0E 04 0.1537E 01 0.1162E C2 0.0            0.3790E 01 0.2871E 01
PL-239 C.2150E 04 C.1159E 01 0.1164E C2 0.C            0.2775E 01 0.2871E 01
PL-239 C.4650E 04 0.9470E 00 0.1158E 02 0.1689E-01     0.2274E 01 0.2871E 01
PL-239 C.1CC0E 05 0.7470E 00 0.1143E 02 0.1718E 0C     0.1918E 01 0.2872E 01
PL-239 C.2150E 05 C.4666E 00 0.1118E C2 0.2273E 0C     0.1621E 01 0.2875E 01
PL-239 C.4650E 05 C.2377E 00 C.1141E C2 0.2686E 00     0.1549E 01 0.2880E 01
PL-239 C.1CC0E 06 0.2063E 00 0.9262E C1 0.3673E 00     0.1511E 01 0.2891E 01
PL-239 C.2CC0E 06 C.1609E 00 0.7690E 01 0.5234E 00     0.1486E 01 0.2911E 01
PL-239 0.4CC0E 06 0.9632E-01 C.5696E 01 0.7326E 00     0.1598E 01 0.2952E 01
PL-239 C.8CC0E 06 0.4315E-01 C.4495E 01 0.8483E 00     0.1762E 01 0.3018E 01
PL-239 C.14C0E 07 0.2736E-02 0.4304E 01 0.1068E 01     0.1953E 01 0.3128E 01
PL-239 C.25C0E 07 0.0        0.4551E C1 0.1227E 01     0.1993E 01 0.3292E 01
PL-239 C.4CC0E 07 0.0        0.4468E C1 0.1460E 01     0.1894E 01 0.3532E 01
PL-239 C.65C0E 07 0.0        0.3676E 01 0.1183E 01     0.2225E 01 0.3911E 01
PL-239 C.21C0E-01     0       0
PL-239 C.215CE 00     0       0
PL-239 C.4650E 00     C       0
PL-239 C.1CC0E 01     C       0
PL-239 C.2150E 01     0       0
PL-239 C.4650E 01     0       0
PL-239 C.1CC0E 02     C       0
PL-239 C.2150E 02     C       0
```

Input data of Sample Problem No. 1 (LTW-Program), continued

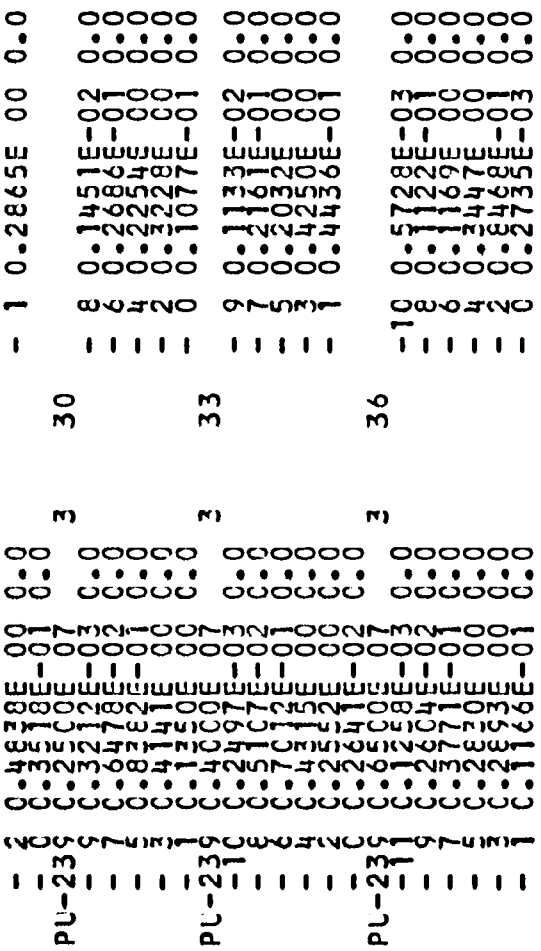
Input data of Sample Problem No. 1 (LTW-Program), continued



Input data of Sample Problem No. 1 (LTW-Program), continued



Input data of Sample Problem No. 1 (LTW-Program), continued



The input data consist of a library which contains a 26 group cross section set of Pu-239 and C12. The C12 data are used in the third sample case (See input 8.1).

4th link: requires the FORTRAN 2, Version 2 monitor and consists of the binary deck LOADER. It uses the "System Tape No. 1" on A5 and a Re-Run Tape on B5.

The input data consist of the specification of monitors and regions (Input 9.1) used by the NDP program:

```
     1
* JEZEBEL, 26 GROUPS - SAMPLE PROBLEM NO. 1
   3
   1   1   2   2   1   0
PU-239
   1   1
PU-239       0.0
   2   1
PU-239  0.038952
   0   2
   1   2
     1
    10      1.0E03      10.0E06 0.965OE06  2.290OE-6  4.527
```

and the specifications (options), history number, geometry data etc. (Input 10.1) used by the RWS program:

```
     2
* JEZEBEL, 26 GROUPS - SAMPLE PROBLEM NO. 1
       BCD 6LEAK *TRAN *MTIM *ITER *FLUX *ENDE *
       TRA 4,4
       DEC 20000,20000
       TRA 4,4
       DEC 20,,1,0,6.284,20.,20.,20.
       TRA 4,4
       DEC 8,0.3981E-5,1.0E-6,0.3981E-6,1.0E-7,0.3981E-7,1.0E-8
       DEC 0.3981E-1.0E-9
       TRA 3,4
```

The output reads:

```
CODE = TIMCC (H.RIEF) *** NUCLEAR DATA PREPARATION (N.D.P.) PROGRAM
* JEZEBEL, 26 GRCUPS - SAMPLE PROBLEM NO. 1
ISCTCPE NO. 1 = PU-239

MIXT. NO. 1
  PU-239-DENS.= 0.          ,
MIXT. NO. 2
  PU-239-DENS.= 0.38952E-01,

RG  0 MIXCT NO. 2
RG  1 MIXCT NO. 2


FISS.-SPECT. PARAMETERS FOR PRIM.NEUTRONS
FISS.SPEC.=TYPE NO    10 FROM E= 0.1000E 04 TO  0.1000E 08(EV), 1)
INPUT DATA ARE WRITTEN ON TAPE 10
```

1) PARAM.= 0.9650E 06, 0.2290E-05, 0.4527E 01

```
TIMCC (H.RIEF, H.KSCHWENDT, EURATOM ISPRA) *** * JEZEBEL, 26 GROUPS - SAMPLE PROBLEM NO. 1
SPECIFICATIONS      LEAK *TRAN *MTIM *ITER *FLUX *ENDE *

THE MACHINE IS FREE FROM LOC. 18117 TO LOC. 32537
SPHERE GEOMETRY *** DATA
NAME SORG LMAX
 20    0    1
LATTICE CONSTANTS    A1= 0.2000000E 02, A2= 0.2000000E 02, A3= 0.2000000E 02
RADII
 0.              0.6284000E 01
SOURCE REGION=   1
```

```
TIMCC (H.BRIEF, F.KSCHWENDT, EURATOM ISPRA) *** * JEZEBEL, 26 GROUPS - SAMPLE PROBLEM NO. 1
NUMB. OF PRIM. HISTORIES =    263, GEN. SECOND.=  20198 (  20198), CALC. SECOND.=  20000 (  20000)
TOTAL LEAKAGE .=  0.672384E 00, UNCOLLI.LEAKAGE= 0.305590E-00
PROBABLE ERROR =  0.134032E-02                   0.21977E-02
SLOWING DOWN DENS.=  0.          , PROB. ERROR =  0.
EMIN=  0.21000E-01  FERMI AGE=     0.      ( AGE(X)=     0.      AGE(Y)=     0.      AGE(Z)=     0.      )
NUCL.NUMB.K=  1  PRODUCTION/SOURCE = 0.992284E 00
K(EFF-COLL.) =  0.99239  0.00046, K(EFF.-FLUX) =  1.00294  0.00275    K(EFFECTIVE) =  1)
NUCL.NUMB.K=  1  MEAN PROD. TIME =  0.28617E-08  0.23521E-10
MEAN PRODUCTION TIME FOR SECONDARY NEUTRONS =  0.28617E-08  0.23521E-10
MEAN DESTRUCT.T= 0.36732E-08, MEAN ABSO. TI. = 0.31108E-08, MEAN LEAKAGE T.= 0.30840E-08,
PROBABLE ERROR =  0.25152E-10                  0.26267E-10                  0.24502E-10

        TIME INTERVAL          PRD.TI.DI.*DT  DST.TI.DI.*DT  PRD.TI.DI.     DST.TI.DI.     2)
  0.          -1.0000E-09  0.          0.          0.E-09     0.E-09
 1.0000E-09 -1.3981E-08  0.3332E-00  0.3347E-00  0.3332E-09  0.3347E-09
 1.3981E-08 -1.9553E-08  0.1493E-01  0.1515E-01  0.1068E-08  0.1083E-08
 1.9553E-08 -2.7344E-07  0.4371E-01  0.4422E-01  0.1466E-07  0.1483E-07
 2.7344E-07 -1.0000E-06  0.2274E-02  0.2301E-02  0.3779E-05  0.3823E-05
 1.0000E-06 -1.3981E-06  0.1638E-03  0.1642E-03  0.5664E-03  0.6189E-03
 1.3981E-06 -1.9553E-06  0.          0.1193E-06  0.          0.1983E-05
 1.9553E-06 -2.7344E-05  0.          0.          0.          0.
 2.7344E-05     AND MORE  0.          0.

RG= 1  NUCL.NO K= 1 FISSIONS PER NEUTRON = 0.320627E-00

LAST USED RANDOMNUMBER =  28367864715
```

1) K(EFFECTIVE) = 1.00098 0.00266

2)

CO.TO PRD.T.	CO.TO DST.T.
0.1667E-09	0.1227E-09
0.9336E-09	0.1059E-08
0.9933E-09	0.1304E-08
0.7094E-09	0.9845E-09
0.1244E-09	0.1371E-09
0.2424E-10	0.2559E-10
0.	0.4878E-13
0.	0.
0.	

TIMCC (H.RIEF, H.KSCHWENDT, EURATOM ISPRA) *** * JEZEBEL, 26 GROUPS - SAMPLE PROBLEM NO. 1

RG	ENERGY GROUP	ABSORPTION	ABSORP./VOL	FLUX(E)*DE	FLUXE(E)
1	0.4650E 02-0.1000E 03	0.19396E-04	0.18660E-07	0.35440E-08	0.66244E-10
1	0.1000E 04-0.2150E 04	0.19355E-03	0.18621E-06	0.89740E-06	0.78035E-09
1	0.2150E 04-0.4650E 04	0.20477E-03	0.19700E-06	0.12856E-05	0.51424E-09
1	0.4650E 04-0.1000E 05	0.37158E-03	0.35749E-06	0.28493E-05	0.53258E-09
1	0.1000E 05-0.2150E 05	0.68993E-03	0.66376E-06	0.63942E-05	0.55601E-09
1	0.2150E 05-0.4650E 05	0.14851E-02	0.14287E-05	0.17570E-04	0.70281E-09
1	0.4650E 05-0.1000E 06	0.43245E-02	0.41605E-05	0.59780E-04	0.11174E-08
1	0.1000E 06-0.2000E 06	0.11364E-01	0.10933E-04	0.16344E-03	0.16344E-08
1	0.2000E 06-0.4000E 06	0.27017E-01	0.25992E-04	0.40518E-03	0.20259E-08
1	0.4000E 06-0.8000E 06	0.55646E-01	0.53535E-04	0.81117E-03	0.20279E-08
1	0.8000E 06-0.1400E 07	0.65186E-01	0.62713E-04	0.89189E-03	0.14865E-08
1	0.1400E 07-0.2500E 07	0.77438E-01	0.74500E-04	0.97795E-03	0.88905E-09
1	0.2500E 07-0.4000E 07	0.52290E-01	0.50306E-04	0.64801E-03	0.43201E-09
1	0.4000E 07-0.6500E 07	0.27069E-01	0.26042E-04	0.35300E-03	0.14120E-09
1	0.6500E 07-0.1050E 08	0.74968E-02	0.72123E-05	0.83218E-04	0.20804E-10
1	0.2100E-01-0.1050E 08	0.33080E-00	0.31825E-03	0.44226E-02	

1)

TOTAL ABSO.(FLUX) = 0.33080E-00, PROB.ERROR = 0.90078E-03
TOTAL ABSO.(COLL.)= 0.32765E-00, PROB.ERROR = 0.13403E-02

TOTAL ABSORPTION = 0.33029E-00, PROB.ERROR = 0.87887E-03

1)

NORM.-FLX(E)	FLUX(E)*E
0.14410E-10	0.48524E-08
0.16975E-09	0.12290E-05
0.11186E-09	0.17484E-05
0.11585E-09	0.39012E-05
0.12095E-09	0.87572E-05
0.15288E-09	0.23896E-04
0.24307E-09	0.81849E-04
0.35553E-09	0.24516E-03
0.44070E-09	0.6 777E-03
0.44114E-09	0.12767E-02
0.32336E-09	0.16351E-02
0.19340E-09	0.17336E-02
0.93975E-10	0.14040E-02
0.30715E-10	0.74130E-03
0.45256E-11	0.17684E-03

TIMCC (H.RIEF, H.KSCHWENDT, EURATOM ISPRA) *** * JEZEBEL, 26 GROUPS - SAMPLE PROBLEM NO. 1

RADIUS	RG	ENERGY GROUP	TRA(IN/OUT)	TRA(OUT/IN)	TRA(SM)	CUR(IN/OUT)
6.284	1	0.1000E 04-0.2150E 04	0.12060E-06	0.	0.12060E-06	0.94935E-07
6.284	1	0.2150E 04-0.4650E 04	0.23039E-06	0.	0.23039E-06	0.12480E-06
6.284	1	0.4650E 04-0.1000E 05	0.81702E-06	0.	0.81702E-06	0.58798E-06
6.284	1	0.1000E 05-0.2150E 05	0.17945E-05	0.	0.17945E-05	0.12946E-05
6.284	1	0.2150E 05-0.4650E 05	0.35777E-05	0.	0.35777E-05	0.25293E-05
6.284	1	0.4650E 05-0.1000E 06	0.12508E-04	0.	0.12508E-04	0.91373E-05
6.284	1	0.1000E 06-0.2000E 06	0.42780E-04	0.	0.42780E-04	0.32167E-04
6.284	1	0.2000E 06-0.4000E 06	0.11463E-03	0.	0.11463E-03	0.85087E-04
6.284	1	0.4000E 06-0.8000E 06	0.24458E-03	0.	0.24458E-03	0.18455E-03
6.284	1	0.8000E 06-0.1400E 07	0.27506E-03	0.	0.27506E-03	0.20734E-03
6.284	1	0.1400E 07-0.2500E 07	0.30350E-03	0.	0.30350E-03	0.23051E-03
6.284	1	0.2500E 07-0.4000E 07	0.21383E-03	0.	0.21383E-03	0.16405E-03
6.284	1	0.4000E 07-0.6500E 07	0.11292E-03	0.	0.11292E-03	0.86441E-04
6.284	1	0.6500E 07-0.1050E 08	0.28543E-04	0.	0.28543E-04	0.21918E-04
6.284	1	0.2100E-01-0.1050E 08	0.13549E-02	0.	0.13549E-02	0.10258E-02

RADIUS	RG	ENERGY GROUP	TRA(IN/OUT)	TRA(OUT/IN)	TRA(SM)	CUR(IN/OUT)
PER UNIT ENERGY						
6.284	1	0.1000E 04-0.2150E 04	0.10487E-09	0.	0.10487E-09	0.82553E-10
6.284	1	0.2150E 04-0.4650E 04	0.92157E-10	0.	0.92157E-10	0.49922E-10
6.284	1	0.4650E 04-0.1000E 05	0.15271E-09	0.	0.15271E-09	0.10990E-09
6.284	1	0.1000E 05-0.2150E 05	0.15604E-09	0.	0.15604E-09	0.11258E-09
6.284	1	0.2150E 05-0.4650E 05	0.14311E-09	0.	0.14311E-09	0.10117E-09
6.284	1	0.4650E 05-0.1000E 06	0.23379E-09	0.	0.23379E-09	0.17079E-09
6.284	1	0.1000E 06-0.2000E 06	0.42780E-09	0.	0.42780E-09	0.32167E-09
6.284	1	0.2000E 06-0.4000E 06	0.57316E-09	0.	0.57316E-09	0.42543E-09
6.284	1	0.4000E 06-0.8000E 06	0.61144E-09	0.	0.61144E-09	0.46138E-09
6.284	1	0.8000E 06-0.1400E 07	0.45843E-09	0.	0.45843E-09	0.34557E-09
6.284	1	0.1400E 07-0.2500E 07	0.27591E-09	0.	0.27591E-09	0.20956E-09
6.284	1	0.2500E 07-0.4000E 07	0.14256E-09	0.	0.14256E-09	0.10936E-09
6.284	1	0.4000E 07-0.6500E 07	0.45170E-10	0.	0.45170E-10	0.34576E-10
6.284	1	0.6500E 07-0.1050E 08	0.71357E-11	0.	0.71357E-11	0.54794E-11

TIMCC (H.RIEF, H.KSCHWENDT, EURATOM ISPRA) *** * JEZEBEL, 26 GROUPS - SAMPLE PROBLEM NO. 1

RG	ENERGY GROUP		EN.-DEPOS.	EN.-DEP./V0	ED/(V0*DE)
0	0.	0.6500E 07	0.	0.	
1	0.	-0.2100E-01	0.88837E-09	0.85467E-12	0.40699E-10
1	0.2100E-01	-0.2150E-00	0.12334E-04	0.11866E-07	0.61165E-07
1	0.2150E-00	-0.4650E-00	0.74259E-04	0.71442E-07	0.28577E-06
1	0.4650E-00	-0.1000E 01	0.25831E-03	0.24851E-06	0.46450E-06
1	0.1000E 01	-0.2150E 01	0.71410E-03	0.68700E-06	0.59739E-06
1	0.2150E 01	-0.4650E 01	0.25025E-02	0.24075E-05	0.96301E-06
1	0.4650E 01	-0.1000E 02	0.15706E-01	0.15110E-04	0.28243E-05
1	0.1000E 02	-0.2150E 02	0.66506E-01	0.63983E-04	0.55637E-05
1	0.2150E 02	-0.4650E 02	0.35609E-00	0.34258E-03	0.13703E-04
1	0.4650E 02	-0.1000E 03	0.15215E 01	0.14638E-02	0.27360E-04
1	0.1000E 03	-0.2150E 03	0.62546E 01	0.60174E-02	0.52325E-04
1	0.2150E 03	-0.4650E 03	0.27615E 02	0.26568E-01	0.10627E-03
1	0.4650E 03	-0.1000E 04	0.99618E 02	0.95838E-01	0.17914E-03
1	0.1000E 04	-0.2160E 04	0.30754E 03	0.29587E-00	0.25728E-03
1	0.2150E 04	-0.4650E 04	0.61312E 03	0.58985E 00	0.23594E-03
1	0.4650E 04	-0.1000E 05	0.81547E 03	0.78453E 00	0.14664E-03
1	0.1000E 05	-0.2150E 05	0.12099E 04	0.11640E 01	0.10122E-03
1	0.2150E 05	-0.4650E 05	0.10462E 04	0.10065E 01	0.40261E-04
1	0.4650E 05	-0.1000E 06	0.37730E 03	0.36298E-00	0.67847E-05
1	0.1000E 06	-0.2000E 06	0.31652E 02	0.30451E-01	0.30451E-06
1	0.	-0.2000E 06	0.45366E 04	0.43645E 01	

TIMCC (H.RIEF, H.KSCHWENDT, EURATOM ISPRA) *** * JEZEBEL, 26 GROUPS - SAMPLE PROBLEM NO. 1

ENERGY GROUP	NO.NTR.	NO.COLLS.	CHI-PRIM*DE	CHI-SEC*DE	CHI-PRIM	CHI-SEC
0.2100E-01-0.2150E-00	0	0	0.	0.	0.	0.
0.2150E-00-0.4650E-00	0	0	0.	0.	0.	0.
0.4650E-00-0.1000E 01	0	0	0.	0.	0.	0.
0.1000E 01-0.2150E 01	0	0	0.	0.	0.	0.
0.2150E 01-0.4650E 01	0	0	0.	0.	0.	0.
0.4650E 01-0.1000E 02	0	0	0.	0.	0.	0.
0.1000E 02-0.2150E 02	0	0	0.	0.	0.	0.
0.2150E 02-0.4650E 02	0	0	0.	0.	0.	0.
0.4650E 02-0.1000E 03	1	4	0.	0.50000E-04	0.	0.93458E-06
0.1000E 03-0.2150E 03	0	0	0.	0.	0.	0.
0.2150E 03-0.4650E 03	0	0	0.	0.	0.	0.
0.4650E 03-0.1000E 04	0	0	0.	0.	0.	0.
0.1000E 04-0.2150E 04	5	40	0.	0.25000E-03	0.	0.21739E-06
0.2150E 04-0.4650E 04	7	39	0.	0.35000E-03	0.	0.14000E-06
0.4650E 04-0.1000E 05	15	85	0.	0.75000E-03	0.	0.14019E-06
0.1000E 05-0.2150E 05	24	83	0.	0.12000E-02	0.	0.10435E-06
0.2150E 05-0.4650E 05	51	285	0.98522E-02	0.24500E-02	0.39409E-06	0.98000E-07
0.4650E 05-0.1000E 06	150	743	0.49261E-02	0.74500E-02	0.92077E-07	0.13925E-06
0.1000E 06-0.2000E 06	465	1791	0.34483E-01	0.22900E-01	0.34483E-06	0.22900E-06
0.2000E 06-0.4000E 06	1196	3448	0.54187E-01	0.59250E-01	0.27094E-06	0.29625E-06
0.4000E 06-0.8000E 06	2746	5900	0.12808E-00	0.13600E-00	0.32020E-06	0.34000E-06
0.8000E 06-0.1400E 07	3906	7440	0.19704E-00	0.19330E-00	0.32841E-06	0.32217E-06
0.1400E 07-0.2500E 07	5289	10432	0.27094E-00	0.26170E-00	0.24631E-06	0.23791E-06
0.2500E 07-0.4000E 07	3780	7319	0.20690E-00	0.18690E-00	0.13793E-06	0.12460E-06
0.4000E 07-0.6500E 07	2064	4202	0.78818E-01	0.10240E-00	0.31527E-07	0.40960E-07
0.6500E 07-0.1050E 08	504	882	0.14778E-01	0.25050E-01	0.36946E-08	0.62625E-08
0.2100E-01-0.1050E 08	20203	42693	1.00000E 00	1.00000E 00		

END OF JOB

12.2 Calculation of Differential Effects in an Assembly of which a Small Piece had been removed

In this example a nearly critical assembly consisting of drawers of fuel elements is analysed. As shown in Fig. 6 the removal worth of a fuel drawer in the perturbed assembly had been studied. For this purpose so-called "small effect calculations" (Sect. 6.3) had to be performed for the unperturbed and the perturbed assembly.

1st link: idential to the previous example; uses "System Tape No. 2".

2nd link: requires the FORTRAN 2, Version 2 monitor and consists of the binary decks, SPACE, BEGIN, WPK, RWS, CORTIMG1, 9EX203, 9TSH, 9STH, 9RER, 9WER, 9IOH, 9IOU, 9EXEM, 9EXIT, 9TES, 9SORT, GEOM, JOM4 til JOM10, JOM13, JOM14, LOC, OMJ8, LOOK7, GPATH, PATH1, START, RAND, ENTRY, WRTP, BVTBF, GREAD, RSTRT, JOMIN, JOM11, JOM12, JOM16, JOM17, END, PTW. Uses the System Tape No. 2 on A5.

Input:

```
2                    (in column 6)
```

Output:

```
TIMOC RWS PROGRAM IS WRITTEN ON TAPE A 5, FILE 2, GEOM. = GEOM
```

3rd link: requires the FORTRAN 2, Version 2 or the FORTRAN 3 monitor and consists of the binary decks: LTW, LTWF. It uses the "System Tape No. 2" on A5.

In this example a 2 group cross section set is used in order to make the results comparable with equivalent calculations performed by a low-order discrete ordinates model.

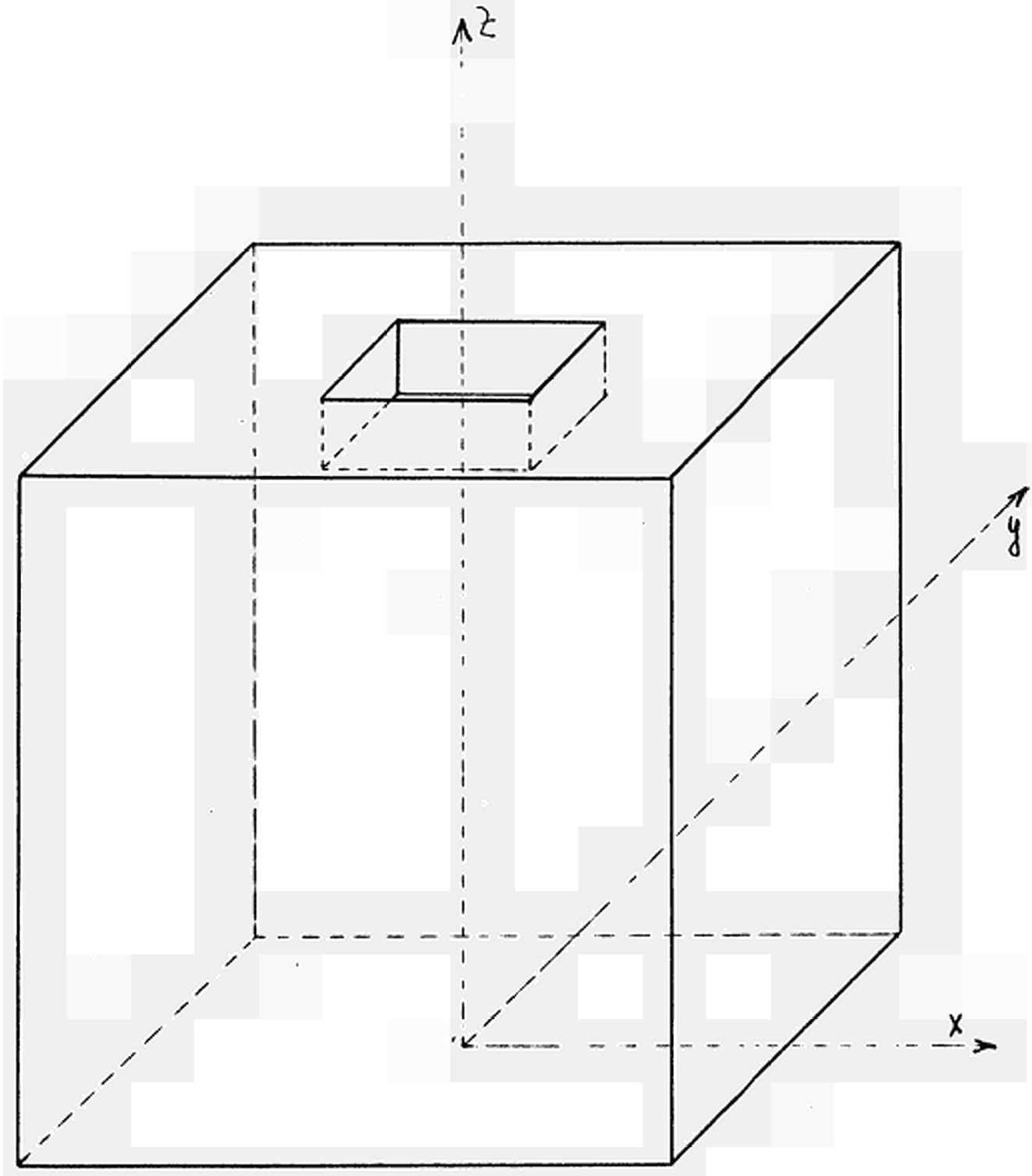


Fig. 6: In this nearly cubical fuel assembly the differential effects (such as: Δk_{eff}, perturbations of flux spectra etc.) caused by the removal of the indicated (small) section are calculated by the use of the SMEC * option.

```
    S           3
    N
    2
C.0
C.3928E 06
1.0    E 06
FUEL
TWO GROUP CROSS-SECTION DATA FOR THE CALCULATION OF A FAST REACTOR

FUEL                1
    1C  0.C         1.CCCCE C6 1.CCCCE 26 1.CCCCE-10 1.5CCCE 04
FUEL   C.0          C.1456E-01 0.0        4.7364E-01 0.4546E-01 2.43   E 00
FUEL   C.3928E 06   C.1333E-03 0.0        2.4224E-01 0.29CCE-01 2.56   E 00
FUEL   C.0                  3
     C 4.7364E-01 C.3276E CO
FUEL   C.3928E 06           6
     C C.1EC5E 00 C.4811E CO    -1 C.6171E-01 0.2445E OC
FUEL*
THE DATA OF FUEL* ARE IDENTICAL TO FUEL. THE SMEC* OPTION REQUIRES
THAT FISSIONABLE ISOTOPES ARE NAMED DIFFERENTLY IN THE PERTURBED
REGION. SEE TIMOC CODE MANUAL SECTIONS 6.3 AND 10.4.7.
FUEL*               1
    1C  0.C         1.CCCCE C6 1.CCCCE 26 1.CCCCE-10 1.5CCCE 04
FUEL* C.0           C.1456E-01 0.0        4.7364E-01 0.4546E-01 2.43   E 00
FUEL* C.3928E 06    C.1333E-03 0.0        2.4224E-01 0.29CCE-01 2.56   E 00
FUEL* C.0                   3
     C 4.7364E-01 C.3276E CO
FUEL* C.3928E 06            6
     C C.1EC5E 00 0.4811E 00    -1 0.6171E-01 0.2445E OC
```

4th link: requires the FORTRAN 2, Version 2 monitor and consists of the binary deck LOADER. Uses the "System Tape No. 2" on A5 and a Re-Run Tape on B5.

a) Input data for the unperturbed case (all the volume of the cube filled with fuel):

```
SMALL EFFECT CALCULATION (SMEC * OPTION) - SAMPLE PROB.NO 2
    2   2   3   9        0
 FUEL
 FUEL*
   1   1
 FUEL   0.0
   2   1
 FUEL   1.0
   3   1
 FUEL*  1.0
   0   1
   1   2
   2   3
   3   2
   4   3
   5   2
   6   3
   7   2
   8   2
    1
   10   0.0        1.0000E 06 1.0000E 26 1.0000E-10 1.5000E 04
    2
SMALL EFFECT CALCULATION (SMEC * OPTION) - SAMPLE PROB.NO 2
       BCD SLEAK *FLUX *ITER *SMEC *RLRU *
       TRA 4,4
       DEC 5000,5000
       TRA 4,4
    2      FEMALE SINGLE
X ZONE              -11.,-3.6666667, 3.6666667,        11.
Y ZONE              -11.,        11.
Z ZONE                0.,        23.7
ZONE        1     1     1
X BLOCK             -11.,-3.6666667
Y BLOCK             -11.,        11.
Z BLOCK               0.,        23.7
BLOCK       1     1     1
MEDIA          1
ZONE        2     1     1
X BLOCK     -3.6666667, 3.6666667
Y BLOCK             -11.,-3.6666667, 3.6666667,        11.
Z BLOCK               0.,21.0666667,        23.7
BLOCK       1     1     1
MEDIA          2
BLOCK       1     2     1
MEDIA          3
BLOCK       1     3     1
MEDIA          4
BLOCK       1     1     2
MEDIA          5
BLOCK       1     2     2
MEDIA          6
BLOCK       1     3     2
MEDIA          7
```

Input data for the "unperturbed case" (Sample Problem No. 2), continued:

```
ZONE        3       1       1
X BLOCK          3.6666667,          11.
Y BLOCK                -11.;          11.
Z BLOCK                   0.;        23.7
BLOCK       1       1       1
MEDIA                   8
     0      NO QUADRIC SURFACE
       7              1              6
 -5.0                0.              10.0          3823.6                 1
  0.0               -5.0             10.0          1132.9                 2
  0.0                0.0             10.0          1132.9                 3
  0.0                5.0             10.0          1132.9                 4
  0.0               -5.0             22.0           141.6                 5
  0.0                5.0             22.0           141.6                 7
  5.0                0.              10.0          3823.6                 8
```

b) Input data for the "perturbed case" (Region 6 without fuel!)

```
      1
SMALL EFFECT CALCULATION (SMEC * CPTION) - SAMPLE PROB.NO 2
      3       2       3       9               0
 FUEL
 FUEL*
        1
 FUEL    0.0
 FUEL    1.0
 FUEL*   1.0
      0
      1
      2
      3
      4
      5
      6
      7
      8
        1
     10   0.0          1.0000E 06 1.0000E 26 1.0000E-10 1.5000E 04
```

Input data for the "perturbed case" (Sample Problem No. 2), continued:

```
2
SMALL EFFECT CALCULATION (SMEC * OPTION) - SAMPLE PROBLEM 2
      BCD SLEAK *FLUX *ITER *SMEC *RLRU *
      TRA 4,4
      DEC 5000,5000
      TRA 4,4
   2     FEMALE SINGLE
X ZONE              -11.,-3.6666667, 3.6666667,        11.
Y ZONE              -11.,        11.
Z ZONE                0.,       23.7
ZONE       1    1    1
X BLOCK             -11.,-3.6666667
Y BLOCK             -11.,        11.
Z BLOCK               0.,       23.7
BLOCK      1    1    1
MEDIA          1
ZONE       2    1    1
X BLOCK    -3.6666667, 3.6666667
Y BLOCK             -11.,-3.6666667, 3.6666667,        11.
Z BLOCK               0.,21.0666667,       23.7
BLOCK      1    1    1
MEDIA          2
BLOCK      1    2    1
MEDIA          3
BLOCK      1    3    1
MEDIA          4
BLOCK      1    1    2
MEDIA          5
BLOCK      1    2    2
MEDIA          6
BLOCK      1    3    2
MEDIA          7
ZONE       3    1    1
X BLOCK     3.6666667,        11.
Y BLOCK             -11.,        11.
Z BLOCK               0.,       23.7
BLOCK      1    1    1
MEDIA          8
   0   NO QUADRIC SURFACE
   7       1       6
 -5.0        0.          10.0        3823.6          1
  0.0       -5.0         10.0        1132.9          2
  0.0        0.0         10.0        1132.9          3
  0.0        5.0         10.0        1132.9          4
  0.0       -5.0         22.0         141.6          5
  0.0        5.0         22.0         141.6          7
  5.0        0.          10.0        3823.6          8
```

c) Results:

The comparison of the two cases (listed below) results in a $k_{eff} = 0.011$, where the method of similar flight paths reduces considerably the error of the difference. The advantage of this method holds of course also for effects which are described by the differences of the other results, like fluxes or absorption ratios.

```
CODE = TIMCC (F.RIEF) *** NUCLEAR DATA PREPARATION (N.D.P.) PROGRAM
SMALL EFFECT CALCULATION (SMEC * OPTION) - SAMPLE PROB.NO 2
ISOTOPE NO. 1 =  FUEL
ISOTOPE NO. 2 =  FUEL*

MIXT. NO. 1
   FUEL -DENS.= 0.          ,
MIXT. NO. 2
   FUEL -DENS.= 0.10000E 01,
MIXT. NO. 3
   FUEL*-DENS.= 0.10000E 01,

RG  0 MIXCT NO. 1
RG  1 MIXCT NO. 2
RG  2 MIXCT NO. 2
RG  3 MIXCT NO. 2
RG  4 MIXCT NO. 2
RG  5 MIXCT NO. 2
RG  6 MIXCT NO. 3
RG  7 MIXCT NO. 2
RG  8 MIXCT NO. 2

FISS.-SPECT. PARAMETERS FOR PRIM.NEUTRONS
FISS.SPEC.*TYPE NO    10 FROM E= 0.          TO  0.1000E 07(EV), 1)
INPUT DATA ARE WRITTEN ON TAPE 10
```

1) PARAM.= 0.1000E 27, 0.1000E-09, 0.1500E 05

```
    2      FEMALE SINGLE
X ZONE          -0.11000E 02, -0.36667E 01,  0.36667E 01,  0.11000E 02
Y ZONE          -0.11000E 02,  0.11000E 02
Z ZONE           0.          ,  0.23700E 02

ZONE       1    1     1
X BLOCK         -0.11000E 02, -0.36667E 01
Y BLOCK         -0.11000E 02,  0.11000E 02
Z BLOCK          0.          ,  0.23700E 02

BLOCK      1    1     1
MEDIA             1

ZONE       2    1     1
X BLOCK         -0.36667E 01,  0.36667E 01
Y BLOCK         -0.11000E 02, -0.36667E 01,  0.36667E 01,  0.11000E 02
Z BLOCK          0.          ,  0.21067E 02,  0.23700E 02

BLOCK      1    1     1
MEDIA             2

BLOCK      1    2     1
MEDIA             3

BLOCK      1    3     1
MEDIA             4

BLOCK      1    1     2
MEDIA             5

BLOCK      1    2     2
MEDIA             6

BLOCK      1    3     2
MEDIA             7

ZONE       3    1     1
X BLOCK          0.36667E 01,  0.11000E 02
Y BLOCK         -0.11000E 02,  0.11000E 02
Z BLOCK          0.          ,  0.23700E 02

BLOCK      1    1     1
MEDIA             8

    0  NO QUADRIC SURFACE
```

THE 1 SMEC-CONNECTED MEDIA ARE 6

RANDOM START

THE PRIMARY NEUTRONS SHALL BE STARTED IN 7 SECTORS
HAVING THE * AND BEING CONTAINED BETWEEN THE FOLLOWING BLOCKBOUNDARIES

MEDIUM *	X-COORDINATES		Y-COORDINATES		Z-COORDINATES	
1	-11.00/	-3.67	-11.00/	11.00	0. /	23.70
2	-3.67/	3.67	-11.00/	-3.67	0. /	21.07
3	-3.67/	3.67	-3.67/	3.67	0. /	21.07
4	-3.67/	3.67	3.67/	11.00	0. /	21.07
5	-3.67/	3.67	-11.00/	-3.67	21.07/	23.70
7	-3.67/	3.67	3.67/	11.00	21.07/	23.70
8	3.67/	11.00	-11.00/	11.00	0. /	23.70

POZZONI - CISANO BERG.

* WEIGHT	BLZON
3823.600	000204100201
1132.900	000204100202
1132.900	000210100402
1132.900	000214100602
141.600	000404101002
141.600	000414101402
3823.600	000204100203

```
TIMOC (H.RIEF, H.KSCHWENDT, EURATOM ISPRA) *** SMALL EFFECT CALCULATION (SMEC * OPTION) - SAMPLE PROB.NO 2
SPECIFICATIONS      LEAK *FLUX *ITER *SMEC *RURU *
RUSS. ROULETTE VERSION, WEIGHT CORRECTION (PRIM) = -0.33465E 01, (SEC.) =  0.37230E 01
THE MACHINE IS FREE FROM LOC. 21585 TO LOC. 32221
GEOMETRY GEOM(CSR) , SEE INPUT PRINT OF RWS.
SOURCE REGION= -C
```

```
TIMCC (H.RIEF, H.KSCHWENDT, EURATOM ISPRA) *** SMALL EFFECT CALCULATION (SMEC * OPTION) - SAMPLE PROB.NO 2
NUMB. OF PRIM. HISTORIES =    257, GEN. SECOND.=   5264 (   5306), CALC. SECOND.=   5000 (   5040)

TOTAL LEAKAGE =  0.52583E 00, UNCOLLI.LEAKAGE= 0.13898E-00
PROBABLE ERROR = 0.38187E-02                    0.32853E-02

SLOWING DOWN DENS.=  0.          , PROB.ERROR =  0.

EMIN=  0.          FERMI AGE=    0.     ( AGE(X)=    0.     AGE(Y)=    0.     AGE(Z)=    0.     )

NUCL.NUMB.K=  1  PRODUCTION/SOURCE = 0.969207E 00
NUCL.NUMB.K=  2  PRODUCTION/SOURCE = 0.879606E-02

K(EFF-COLL.) =  0.97807  0.00786, K(EFF.-FLUX) =  0.98267  0.00718    K(EFFECTIVE) =  0.98173  0.00713

MULTIPLICATION FACTOR IF REGION SPECIFIED BY  SMEC *  IS BLACK

K(EFF-COLL.) =  0.95364  0.00782, K(EFF.-FLUX) =  0.95848  0.00713    K(EFFECTIVE) =  0.95753  0.00709


RG= 1  NUCL.NO K= 1 FISSIONS PER NEUTRON = 0.122270E-00
RG= 2  NUCL.NO K= 1 FISSIONS PER NEUTRON = 0.467552E-01
RG= 3  NUCL.NO K= 1 FISSIONS PER NEUTRON = 0.660364E-01
RG= 4  NUCL.NO K= 1 FISSIONS PER NEUTRON = 0.440692E-01
RG= 5  NUCL.NO K= 1 FISSIONS PER NEUTRON = 0.240447E-02
RG= 6  NUCL.NO K= 2 FISSIONS PER NEUTRON = 0.344078E-02
RG= 7  NUCL.NO K= 1 FISSIONS PER NEUTRON = 0.251649E-02
RG= 8  NUCL.NO K= 1 FISSIONS PER NEUTRON = 0.108883E-00

LAST USED RANDOMNUMBER =   8456701203
```

Output of Sample Problem No. 2 (RWS-Program, "Unperturbed Case"), continued:

1)

TIMOC (H.RIEF, H.KSCHWENDT, EURATOM ISPRA) *** SMALL EFFECT CALCULATION (SMEC * OPTION) -

RG	ENERGY GROUP	ABSORPTION	ABSORP./VOL	FLUX(E)*DE	FLUXE(E)
1	0. -0.3928E 06	0.98603E-01	0.98603E-01	0.16428E 01	0.41824E-05
1	0.3928E 06-0.1000E 07	0.47806E-01	0.47806E-01	0.16409E 01	0.27025E-05
1	0. -0.1000E 07	0.14641E-00	0.14641E-00	0.32838E 01	
2	0. -0.3928E 06	0.39394E-01	0.39394E-01	0.65635E 00	0.16709E-05
2	0.3928E 06-0.1000E 07	0.16995E-01	0.16995E-01	0.58337E 00	0.96075E-06
2	0. -0.1000E 07	0.56389E-01	0.56389E-01	0.12397E 01	
3	0. -0.3928E 06	0.56022E-01	0.56022E-01	0.93339E 00	0.23762E-05
3	0.3928E 06-0.1000E 07	0.23713E-01	0.23713E-01	0.81395E 00	0.13405E-05
3	0. -0.1000E 07	0.79735E-01	0.79735E-01	0.17473E 01	
4	0. -0.3928E 06	0.35156E-01	0.35156E-01	0.58574E 00	0.14912E-05
4	0.3928E 06-0.1000E 07	0.17522E-01	0.17522E-01	0.60143E 00	0.99050E-06
4	0. -0.1000E 07	0.52678E-01	0.52678E-01	0.11872E 01	
5	0. -0.3928E 06	0.19081E-02	0.19081E-02	0.31791E-01	0.80935E-07
5	0.3928E 06-0.1000E 07	0.96365E-03	0.96365E-03	0.33077E-01	0.54475E-07
5	0. -0.1000E 07	0.28718E-02	0.28718E-02	0.64868E-01	
6	0. -0.3928E 06	0.29680E-02	0.29680E-02	0.49450E-01	0.12589E-06
6	0.3928E 06-0.1000E 07	0.11983E-02	0.11983E-02	0.41131E-01	0.67738E-07
6	0. -0.1000E 07	0.41663E-02	0.41663E-02	0.90581E-01	
7	0. -0.3928E 06	0.20437E-02	0.20437E-02	0.34051E-01	0.86687E-07
7	0.3928E 06-0.1000E 07	0.97299E-03	0.97299E-03	0.33398E-01	0.55003E-07
7	0. -0.1000E 07	0.30167E-02	0.30167E-02	0.67449E-01	
8	0. -0.3928E 06	0.89566E-01	0.89566E-01	0.14923E 01	0.37920E-05
8	0.3928E 06-0.1000E 07	0.41234E-01	0.41234E-01	0.14153E 01	0.23309E-05
8	0. -0.1000E 07	0.13080E-00	0.13080E-00	0.29076E 01	

TOTAL ABSO.(FLUX) = 0.47607E-00, PROB.ERROR = 0.38286E-02
TOTAL ABSO.(COLL.)= 0.47401E-00, PROB.ERROR = 0.41081E-02

TOTAL ABSORPTION = 0.47563E-00, PROB.ERROR = 0.38053E-02

Output of Sample Problem No. 2 (RWS-Program, "Unperturbed Case"), continued:

TIMCC (H.RIEF, H.KSCHWENDT, EURATOM ISPRA) *** SMALL EFFECT CALCULATION (SMEC * OPTION) - SAMPLE PROB.NO 2

ENERGY GROUP		NO.NTR.	NO.COLLS.	CHI-PRIM*DE	CHI-SEC*DE	CHI-PRIM	CHI-SEC
0.	-0.3928E 06	1309	9107	0.24903E-00	0.24702E-00	0.63398E-06	0.62888E-06
0.3928E 06	-0.1000E 07	3988	20327	0.75097E 00	0.75298E 00	0.12368E-05	0.12401E-05
0.	-0.1000E 07	5297	29434	1.00000E 00	1.00000E 00		

SAMPLE PROB.NO 2

NORM.-FLX(E)	FLUX(E)
0.39499E-06	0.82141
0.25523E-06	0.18820
0.15781E-06	0.32817
0.90735E-07	0.66907
0.22442E-06	0.46669
0.12660E-06	0.93353
0.14083E-06	0.29287
0.93545E-07	0.68978
0.76437E-08	0.15896
0.51447E-08	0.37936
0.11889E-07	0.24725
0.63973E-08	0.47173
0.81869E-08	0.17025
0.51946E-08	0.38304
0.35879E-06	0.74613
0.22014E-06	0.16233

1)

```
CODE = TIMCC (F.RIEF) *** NUCLEAR DATA PREPARATICN (N.D.P.) PROGRAM
SMALL EFFECT CALCULATION (SMEC * OPTION) - SAMPLE PROB.NO 2
ISCTOPE NO. 1 =  FUEL
ISCTOPE NO. 2 =  FUEL*

MIXT. NO. 1
   FUEL -DENS.= 0.          ,
MIXT. NO. 2
   FUEL -DENS.= 0.1CCOOE 01,
MIXT. NO. 3
   FUEL*-DENS.= 0.1CCOOE 01,

RG  0 MIXCT NO. 1
RG  1 MIXCT NO. 2
RG  2 MIXCT NO. 2
RG  3 MIXCT NO. 2
RG  4 MIXCT NO. 2
RG  5 MIXCT NO. 2
RG  6 MIXCT NO. 1
RG  7 MIXCT NO. 2
RG  8 MIXCT NO. 2

FISS.-SPECT. PARAMETERS FOR PRIM.NEUTRONS
FISS.SPEC.*TYPE NO    10 FROM E= 0.          TO  0.1000E 07(EV), 1)
INPUT DATA ARE WRITTEN ON TAPE 10
```

1) PARAM.= 0.1000E 27, 0.1000E-09, 0.1500E 05

```
    2       FEMALE SINGLE
X ZONE              -0.11000E 02, -0.36667E 01,  0.36667E 01,  0.11000E 02
Y ZONE              -0.11000E 02;  0.11000E 02
Z ZONE               0.          ;  0.23700E 02

ZONE        1    1     1
X BLOCK             -0.11000E 02, -0.36667E 01
Y BLOCK             -0.11000E 02;  0.11000E 02
Z BLOCK              0.          ;  0.23700E 02

BLOCK       1    1     1
MEDIA             1

ZONE        2    1     1
X BLOCK             -0.36667E 01,  0.36667E 01
Y BLOCK             -0.11000E 02; -0.36667E 01,  0.36667E 01,  0.11000E 02
Z BLOCK              0.          ;  0.21067E 02,  0.23700E 02

BLOCK       1    1     1
MEDIA             2

BLOCK       1    2     1
MEDIA             3

BLOCK       1    3     1
MEDIA             4

BLOCK       1    1     2
MEDIA             5

BLOCK       1    2     2
MEDIA             6

BLOCK       1    3     2
MEDIA             7

ZONE        3    1     1
X BLOCK              0.36667E 01,  0.11000E 02
Y BLOCK             -0.11000E 02;  0.11000E 02
Z BLOCK              0.          ;  0.23700E 02

BLOCK       1    1     1
MEDIA             8

    0   NO QUADRIC SURFACE
```

THE 1 SMEC-CONNECTED MEDIA ARE 6

RANCOM START

THE PRIMARY NEUTRONS SHALL BE STARTED IN 7 SECTORS
HAVING THE * AND BEING CONTAINED BETWEEN THE FOLLOWING BLOCKBOUNDARIES

MEDIUM *	X-COORDINATES	Y-COORDINATES	Z-COORDINATES
1	-11.00/ -3.67	-11.00/ 11.00	0. / 23.70
2	-3.67/ 3.67	-11.00/ -3.67	0. / 21.07
3	-3.67/ 3.67	-3.67/ 3.67	0. / 21.07
4	-3.67/ 3.67	3.67/ 11.00	0. / 21.07
5	-3.67/ 3.67	-11.00/ -3.67	21.07/ 23.70
7	-3.67/ 3.67	3.67/ 11.00	21.07/ 23.70
8	3.67/ 11.00	-11.00/ 11.00	0. / 23.70

* WEIGHT	BLZON
3823.600	000204100201
1132.900	000204100202
1132.900	000210100402
1132.900	000214100602
141.600	000404101002
141.600	000414101402
3823.600	000204100203

Output of Sample Problem No. 2 (RWS-Program, "Perturbed Case"), continued:

```
TIMCC (H.RIEF, H.KSCHWENDT, EURATOM ISPRA) *** SMALL EFFECT CALCULATION (SMEC * OPTION) - SAMPLE PROB.NO 2
SPECIFICATIONS      LEAK *FLUX *ITER *SMEC *RURU *
RUSS. ROULETTE VERSION, WEIGHT CORRECTION (PRIM) = -0.33465E 01, (SEC.) =  0.37230E 01
THE MACHINE IS FREE FROM LOC. 21585 TO LOC. 32221
GEOMETRY GEOM(C5R) , SEE INPUT PRINT OF RWS.
SOURCE REGION= -C
```

```
TIMOC (H.RIEF, H.KSCHWENDT, EURATOM ISPRA) *** SMALL EFFECT CALCULATION (SMEC * OPTION) - SAMPLE PROB.NO 2
NUMB. OF PRIM. HISTORIES =    257, GEN. SECOND.=   5264 (   5264), CALC. SECOND.=   5000 (   5000)
TOTAL LEAKAGE =  0.53155E 00, UNCOLLI.LEAKAGE= 0.14129E-00
PROBABLE ERROR = 0.38371E-02                    0.33212E-02
SLOWING DOWN DENS.=  0.          , PROB.ERROR =  0.
EMIN=  0.          FERMI AGE=    0.     ( AGE(X)=    0.     AGE(Y)=    0.     AGE(Z)=    0.     )
NUCL.NUMB.K=  1  PRODUCTION/SOURCE = 0.966754E 00
NUCL.NUMB.K=  2  PRODUCTION/SOURCE = 0.
K(EFF-COLL.) =  0.96681  0.00787, K(EFF.-FLUX) =  0.97150  0.00717    K(EFFECTIVE) =  0.97058  0.00713
MULTIPLICATION FACTOR IF REGION SPECIFIED BY  SMEC *  IS BLACK
K(EFF-COLL.) =  0.96126  0.00784, K(EFF.-FLUX) =  0.96614  0.00714    K(EFFECTIVE) =  0.96518  0.00710

RG= 1  NUCL.NO K= 1 FISSIONS PER NEUTRON = 0.122041E-00
RG= 2  NUCL.NO K= 1 FISSIONS PER NEUTRON = 0.466195E-01
RG= 3  NUCL.NO K= 1 FISSIONS PER NEUTRON = 0.655435E-01
RG= 4  NUCL.NO K= 1 FISSIONS PER NEUTRON = 0.442195E-01
RG= 5  NUCL.NO K= 1 FISSIONS PER NEUTRON = 0.216117E-02
RG= 7  NUCL.NO K= 1 FISSIONS PER NEUTRON = 0.244312E-02
RG= 8  NUCL.NO K= 1 FISSIONS PER NEUTRON = 0.108796E-00

LAST USED RANDOMNUMBER =   8456701203
```

TIMCC (H.RIEF, H.KSCHWENDT, EURATOM ISPRA) *** SMALL EFFECT CALCULATION (SMEC * OPTION) -

RG	ENERGY GROUP	ABSORPTION	ABSORP./VOL	FLUX(E)*DE	FLUXE(E)
1	0. -0.3928E 06	0.98164E-01	0.98164E-01	0.16355E 01	0.41637E-05
1	0.3928E 06-0.1000E 07	0.47909E-01	0.47909E-01	0.16445E 01	0.27083E-05
1	0. -0.1000E 07	0.14607E-00	0.14607E-00	0.32800E 01	
2	0. -0.3928E 06	0.39151E-01	0.39151E-01	0.65230E 00	0.16606E-05
2	0.3928E 06-0.1000E 07	0.17044E-01	0.17044E-01	0.58504E 00	0.96350E-06
2	0. -0.1000E 07	0.56195E-01	0.56195E-01	0.12373E 01	
3	0. -0.3928E 06	0.55477E-01	0.55477E-01	0.92431E 00	0.23531E-05
3	0.3928E 06-0.1000E 07	0.23632E-01	0.23632E-01	0.81110E 00	0.13359E-05
3	0. -0.1000E 07	0.79110E-01	0.79110E-01	0.17355E 01	
4	0. -0.3928E 06	0.35313E-01	0.35313E-01	0.58836E 00	0.14979E-05
4	0.3928E 06-0.1000E 07	0.17553E-01	0.17553E-01	0.60251E 00	0.99227E-06
4	0. -0.1000E 07	0.52866E-01	0.52866E-01	0.11909E 01	
5	0. -0.3928E 06	0.16110E-02	0.16110E-02	0.26940E-01	0.68331E-07
5	0.3928E 06-0.1000E 07	0.94534E-03	0.94534E-03	0.32449E-01	0.53440E-07
5	0. -0.1000E 07	0.25563E-02	0.25563E-02	0.59289E-01	
6	0. -0.3928E 06	0.	0.	0.37384E-01	0.95172E-07
6	0.3928E 06-0.1000E 07	0.	0.	0.38108E-01	0.62761E-07
6	0. -0.1000E 07	0.	0.	0.75492E-01	
7	0. -0.3928E 06	0.19801E-02	0.19801E-02	0.32991E-01	0.83990E-07
7	0.3928E 06-0.1000E 07	0.94768E-03	0.94768E-03	0.32529E-01	0.53572E-07
7	0. -0.1000E 07	0.29278E-02	0.29278E-02	0.65520E-01	
8	0. -0.3928E 06	0.89049E-01	0.89049E-01	0.14837E 01	0.37771E-05
8	0.3928E 06-0.1000E 07	0.41539E-01	0.41539E-01	0.14258E 01	0.23482E-05
8	0. -0.1000E 07	0.13059E-00	0.13059E-00	0.29095E 01	

TOTAL ABSO.(FLUX) = 0.47032E-00, PROB.ERROR = 0.38217E-02
TOTAL ABSO.(COLL.)= 0.46829E-00, PROB.ERROR = 0.41130E-02

TOTAL ABSORPTION = 0.46991E-00, PROB.ERROR = 0.38005E-02

Output of Sample Problem No. 2 (RWS-Program, "Perturbed Case"), continued:

```
TIMCC (H.RIEF, H.KSCHWENDT, EURATOM ISPRA) *** SMALL EFFECT CALCULATION (SMEC * OPTION) - SAMPLE PROB.NO 2

      ENERGY GROUP          NO.NTR.   NO.COLLS.   CHI-PRIM*DE   CHI-SEC*DE    CHI-PRIM      CHI-SEC
 0.          -0.3928E 06      1297        8698   0.24903E-00   0.24660E-00   0.63398E-06   0.62780E-06
0.3928E 06-0.1000E 07         3960       19536   0.75097E 00   0.75340E 00   0.12368E-05   0.12408E-05

 0.          -0.1000E 07      5257       28234   1.00000E 00   1.00000E 00

END OF JOB
```

SAMPLE PROB.NO 2

NORM.-FLX(E)	FLUX(E)
0.39454E-06	0.81776
0.25663E-06	0.18861
0.15735E-06	0.32615
0.91297E-07	0.67098
0.22297E-06	0.46216
0.12659E-06	0.93035
0.14193E-06	0.29418
0.94023E-07	0.69102
0.64747E-08	0.13420
0.50637E-08	0.37216
0.90181E-08	0.18692
0.59469E-08	0.43706
0.79585E-08	0.16496
0.50762E-08	0.37308
0.35790E-06	0.74183
0.22251E-06	0.16353

12.3 Decay of a Neutron Burst in a Graphite Slab

In this example the decay of a monodirectional, monoenergetic, $\delta(t)$-burst is calculated. The neutrons are injected into an infinite graphite slab of 2.54 cm thickness as shown in Fig. 7. The nuclear data consist of a 26 group cross section library generated by the CODAC (26) code. They were already loaded on "System Tape No. 1" file 3 in the first example.

1st link: requires the FORTRAN 2, Version 2 monitor and consists of the binary decks: SPACE, BEGIN, WPK67, RWS, SLBCY, PTW. It uses the "System Tape No. 1" on A5.

Input:

```
4        (in column 6)
```

Output:

```
TIMOC RWS PROGRAM IS WRITTEN ON TAPE A5, FILE 4, GEOM. = SLBCYL
```

2nd link: requires the FORTRAN 2, Version 2 monitor and consists of the library decks: SPACE, BEGIN, WPK67, TTS, PTW. Uses the "System Tape No. 1" on A5.

Input:

```
5        (in column 6)
```

Output:

```
TIMOC TSS PROGRAM IS WRITTEN ON TAPE A5, FILE 5.
```

3rd link: requires the FORTRAN 2, Version 2 monitor and consists of the binary deck LOADER. Uses the "System Tape No. 1" on A5, a Re-

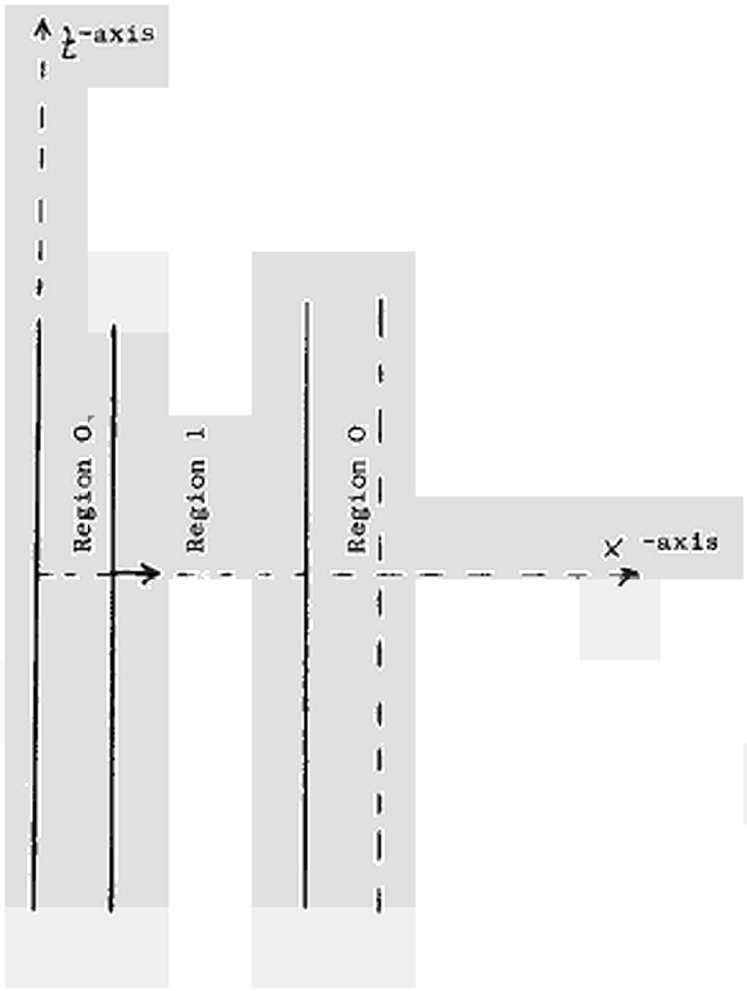


Fig. 7: Monodirectional, monoenergetic, δ (t) source at the boundary of a slab of 2.54 cm [illegible]ness ($\cos\Theta = 1$; $\sin\phi \neq 0$; $\cos\phi = 1$; E_o = 2.1[illegible]).
Region O is the leakage region.

Run Tape on B5 and the Time Scanning Tape on A6.

Input:

```
TIME DEPENDENT NEUTRON DECAY IN A C12-SLAB* SAMPLE PRO.NO.3
C12    0   2   2   1   0
C12    1
C12      0.0803
C12      1
C12    0.0
         2
         1
      1.0E03     10.0E06   0.9650E06  2.2900E-6  4.527
TIME DEPENDENT NEUTRON DECAY IN A C12-SLAB* SAMPLE PRO.NO.3
        BCD 9FLUX *TRAN *LEAK *TIME *STPT *ANGL *STEN *VARC *TMAX *
        TRA 4,4
        DEC 5000,1
        TRA 4,4
        DEC 11,1,1,0,1.0,3.54
        TRA 4,4
        DEC 1.01,0.0,0.0,0.0,1.0,0.0,2.14E6
        TRA 4,4
        DEC 5,5.0E-8,4.0E-8,3.0E-8,2.0E-8,1.0E-8,0.0
        TRA 4,4
        DEC 1,22
        TRA 4,4
TIME DEPENDENT NEUTRON DECAY IN A C12-SLAB* SAMPLE PRO.NO.3          1
        BCD 3FLUX *ENEN *ENGB *
        TRA 4,4
        DEC 5,2.15E6,1.64E6,0.8E6,0.4E6,0.1E6,0.0
        TRA 4,4
        DEC 5,5.0E-8,4.0E-8,3.0E-8,2.0E-8,1.0E-8,0.0
        TRA 4,4
```

Output: In addition to the sampling performed by the RWS program a tape containing the characteristic (time dependent) parameters of all events is generated and afterwards scanned by the Time Tape Scanning (TTS-) program (see 11.7).

```
CODE = TIMOC (H.RIEF) *** NUCLEAR DATA PREPARATION (N.D.P.) PROGRAM
TIME DEPENDENT NEUTRON DECAY IN A C12-SLAB* SAMPLE PRO.NO.3
ISOTOPE NO. 1 = C12

MIXT. NO. 1
  C12    -DENS.= 0.80300E-01,

MIXT. NO. 2
  C12    -DENS.= 0.          ,

RG  0 MIXCT NO. 2
RG  1 MIXCT NO. 1

FISS.-SPECT. PARAMETERS FOR PRIM.NEUTRONS
FISS.SPEC.*TYPE NO    10 FROM E= 0.1000E 04 TO  0.1000E 08(EV), 1)
INPUT DATA ARE WRITTEN ON TAPE 10
```

1) PARAM.= 0.9650E 06, 0.2290E-05, 0.4527E 01

```
TIMOC (H.RIEF, H.KSCHWENDT, EURATOM ISPRA) *** TIME DEPENDENT NEUTRON DECAY IN A C12-SLAB* SAMPLE PRO.NO.3
SPECIFICATIONS      FLUX *TRAN *LEAK *TIME *STPT *ANGL *STEN *VARC *TMAX *

THE MACHINE IS FREE FROM LOC. 18897 TO LOC. 32537
GEOMETRY DATA
TYPE= 111
SLABS
C =  1
R =      0.          1.000      3.540
SOURCE REGION=  1
SPNX=    1.01 SPNY=    0.    SPNZ=    0.    SSINP= 0.          SCOSP= 0.10000E 01 SCOST= 0.    1)

1)  SPNE= 0.21400E 07
```

TIMOC (H.RIEF, H.KSCHWENDT, EURATOM ISPRA) *** TIME DEPENDENT NEUTRON DECAY IN A C12-SLAB* SAMPLE PRO.NO.3

NUMB. OF PRIM. HISTORIES = 5000, GEN. SECOND.= 0 (0), CALC. SECOND.= 0 (0)

TOTAL LEAKAGE = 0.10000E 01, UNCOLLI.LEAKAGE= 0.68260E 00
PROBABLE ERROR = 0. 0.44400E-02

SLOWING DOWN DENS.= 0. , PROB.ERROR = 0.

EMIN= 0.2100E-01 FERMI AGE= 0. (AGE(X)= 0. AGE(Y)= 0. AGE(Z)= 0.)

MEAN DESTRUCT.T= 0.17347E-08, MEAN ABSO. TI. = 0. , MEAN LEAKAGE T.= 0.17347E-08,
PROBABLE ERROR = 0.15189E-10 0. 0.15189E-10

TIME INTERVAL		PRD.TI.DI.*DT	DST.TI.DI.*DT	PRD.TI.DI.	DST.TI.DI. 1)
0.	-1.0000E-08	0.	0.9924E 00	0.	0.9924E 08
1.0000E-08	-0.2000E-07	0.	0.7200E-02	0.	0.7200E 06
0.2000E-07	-0.3000E-07	0.	0.4000E-03	0.	0.4000E 05
0.3000E-07	-0.4000E-07	0.	0.	0.	0.
0.4000E-07	-0.5000E-07	0.	0.	0.	0.

WEIGHT LOST BY EXCEEDING THE TIME LIMIT (0.50000E-07) = 0.

LAST USED RANDOMNUMBER = 20207087649

TOTAL NUMBER OF RECORDS ON TIME TAPE = 126

1)

00.TO PRD.T.	CO.TO DST.T.
0.	0.1632E-08
0.	0.9440E-10
0.	0.8579E-11
0.	0.
0.	0.

TIMOC (H.RIEF, H.KSCHWENDT, EURATOM ISPRA) *** TIME DEPENDENT NEUTRON DECAY IN A C12-SLAB* SAMPLE PRO.NO.3

RG	ENERGY GROUP	ABSORPTION	ABSORP./VOL	FLUX(E)*DE	FLUXE(E) 1)
1	0.4000E 06-0.8000E 06	0.	0.	0.50723E-02	0.12681E-07
1	0.8000E 06-0.1400E 07	0.	0.	0.67042E-01	0.11174E-06
1	0.1400E 07-0.2500E 07	0.	0.	0.12436E 01	0.11305E-05
1	0.2100E-01-0.1050E 08	0.	0.	0.13157E 01	

PROB. ERRORS = 6.380528 PERCENT

1	0.8000E 06-0.1400E 07	0.	0.	0.42777E-02	0.71294E-08

TOTAL ABSO.(FLUX) = 0. , PROB.ERROR = 0.
TOTAL ABSO.(COLL.)= 0. , PROB.ERROR = 0.

TOTAL ABSORPTION = 0. , PROB.ERROR = 0.

1)

NORM.-FLX(E)	FLUX(E)*E
0.37944E-08	0.76084E-02
0.33435E-07	0.12291E-00
0.33829E-06	0.22046E 01

TIMCC (H.RIEF, H.KSCHWENDT, EURATOM ISPRA) *** TIME DEPENDENT NEUTRON DECAY IN A C12-SLAB* SAMPLE PRO.NO.3

RADIUS	RG	ENERGY GROUP	TRA(IN/OUT)	TRA(OUT/IN)	TRA(SM)	CUR(IN/OUT)
1.000	0	0.4000E 06-0.8000E 06	0.	0.50000E-03	0.60000E-03	0.
1.000	0	0.8000E 06-0.1400E 07	0.	0.19000E-01	0.19000E-01	0.
1.000	0	0.1400E 07-0.2500E 07	0.	0.12320E-00	0.12320E-00	0.
1.000	0	0.2100E-01-0.1050E 08	0.	0.14280E-00	0.14280E-00	0.
3.540	1	0.4000E 06-0.8000E 06	0.16000E-02	0.	0.16000E-02	0.10078E-02
3.540	1	0.8000E 06-0.1400E 07	0.20200E-01	0.	0.20200E-01	0.12993E-01
3.540	1	0.1400E 07-0.2500E 07	0.83540E 00	0.	0.83540E 00	0.77626E 00
3.540	1	0.2100E-01-0.1050E 08	0.85720E 00	0.	0.85720E 00	0.79026E 00

1)

RADIUS PER UNIT ENERGY	RG	ENERGY GROUP	TRA(IN/OUT)	TRA(OUT/IN)	TRA(SM)	CUR(IN/OUT)
1.000	0	0.4000E 06-0.8000E 06	0.	0.15000E-08	0.15000E-08	0.
1.000	0	0.8000E 06-0.1400E 07	0.	0.31667E-07	0.31667E-07	0.
1.000	0	0.1400E 07-0.2500E 07	0.	0.11200E-06	0.11200E-06	0.
3.540	1	0.4000E 06-0.8000E 06	0.40000E-08	0.	0.40000E-08	0.25195E-08
3.540	1	0.8000E 06-0.1400E 07	0.33667E-07	0.	0.33667E-07	0.21655E-07
3.540	1	0.1400E 07-0.2500E 07	0.75945E-06	0.	0.75945E-06	0.70569E-06

1)

OUR(OUT/IN)	CUR(SM)
-0.31774E-03	-0.31774E-03
-0.11020E-01	-0.11020E-01
-0.73077E-01	-0.73077E-01
-0.84415E-01	-0.84415E-01
-0.	0.10078E-02
-0.	0.12993E-01
-0.	0.77626E 00
-0.	0.79026E 00

CUR(OUT/IN)	CUR(SM)
-0.79436E-09	-0.79438E-09
-0.18367E-07	-0.18367E-07
-0.66433E-07	-0.66433E-07
-0.	0.25195E-08
-0.	0.21655E-07
-0.	0.70569E-06

TIMOC (H.RIEF, H.KSCHWENDT, EURATOM ISPRA) *** TIME DEPENDENT NEUTRON DECAY IN A C12-SLAB* SAMPLE PRO.NO.3

ENERGY GROUP	NO.NTR.	NO.COLLS.	CHI-PRIM*DE	CHI-SEC*DE	CHI-PRIM	CHI-SEC
0.2100E-01-0.2150E-00	0	0	0.	0.	0.	0.
0.2150E-00-0.4650E-00	0	0	0.	0.	0.	0.
0.4650E-00-0.1000E 01	0	0	0.	0.	0.	0.
0.1000E 01-0.2150E 01	0	0	0.	0.	0.	0.
0.2150E 01-0.4650E 01	0	0	0.	0.	0.	0.
0.4650E 01-0.1000E 02	0	0	0.	0.	0.	0.
0.1000E 02-0.2150E 02	0	0	0.	0.	0.	0.
0.2150E 02-0.4650E 02	0	0	0.	0.	0.	0.
0.4650E 02-0.1000E 03	0	0	0.	0.	0.	0.
0.1000E 03-0.2150E 03	0	0	0.	0.	0.	0.
0.2150E 03-0.4650E 03	0	0	0.	0.	0.	0.
0.4650E 03-0.1000E 04	0	0	0.	0.	0.	0.
0.1000E 04-0.2150E 04	0	0	0.	0.	0.	0.
0.2150E 04-0.4650E 04	0	0	0.	0.	0.	0.
0.4650E 04-0.1000E 05	0	0	0.	0.	0.	0.
0.1000E 05-0.2150E 05	0	0	0.	0.	0.	0.
0.2150E 05-0.4650E 05	0	0	0.	0.	0.	0.
0.4650E 05-0.1000E 06	0	0	0.	0.	0.	0.
0.1000E 06-0.2000E 06	0	0	0.	0.	0.	0.
0.2000E 06-0.4000E 06	0	0	0.	0.	0.	0.
0.4000E 06-0.8000E 06	0	0	0.	0.	0.	0.
0.8000E 06-0.1400E 07	0	0	0.	0.	0.	0.
0.1400E 07-0.2500E 07	5000	2556	0.10000E 01	0.	0.90909E-06	0.
0.2500E 07-0.4000E 07	0	0	0.	0.	0.	0.
0.4000E 07-0.6500E 07	0	0	0.	0.	0.	0.
0.6500E 07-0.1050E 08	0	0	0.	0.	0.	0.
0.2100E-01-0.1050E 08	5000	2556	0.10000E 01	0.		

```
TIMOC ( TTS, BY H.KSCHWENDT, EURATOM ISPRA) ** TIME DEPENDENT NEUTRON DECAY IN A C12-SLAB* SAMPLE PRO.NO.3

SPECIFICATIONS **  FLUX *UNEN *ENGB *
GEOMETRY DATA
TYPE= 111
SLABS
C =  1
R =     0.         1.000      3.540
ENERGY BOUNDARIES, IM=  5
  0.21500E 07  0.16400E 07  0.80000E 06  0.40000E 06  0.10000E 06  0.
TIME BOUNDARIES, TM=  5
  0.50000E-07  0.40000E-07  0.30000E-07  0.20000E-07  1.00000E-08  0.
NUMB. OF PRIM. HISTORIES =   5000, NORMALIZATION=   5000.000, GEN. SEC.=      0
      126 CORRECT RECORDS,        0 WRONG RECORDS
```

TIMOC (TTS, BY H.KSCHWENDT, EURATOM ISPRA) ** TIME DEPENDENT NEUTRON DECAY IN A C12-SLAB* SAMPLE PRO.NO.3

TIME INTERVAL = 0. -1.00000E-08

DISTANCE	CI	ENERGY GROUP	TRA(IN/OUT)	TRA(OUT/IN)	TRA(SM)	FLX(IN/OUT)
1.00	0	0.4000E 06-0.8000E 06	0.	0.20000E 05	0.20000E 05	0.
1.00	0	0.8000E 06-0.1640E 07	0.	0.88200E 07	0.88200E 07	0.
1.00	0	0.1640E 07-0.2150E 07	0.	0.51200E 07	0.51200E 07	0.
1.00	0	0. -0.2150E 07	0.	0.13960E 08	0.13960E 08	0.
3.54	0	0.4000E 06-0.8000E 06	0.60000E 05	0.	0.60000E 05	0.83632E 05
3.54	0	0.8000E 06-0.1640E 07	0.33200E 07	0.	0.33200E 07	0.83998E 07
3.54	0	0.1640E 07-0.2150E 07	0.81900E 08	0.	0.81900E 08	0.10129E 09
3.54	0	0. -0.2150E 07	0.85280E 08	0.	0.85280E 08	0.10977E 09

TIME INTERVAL = 1.00000E-08 -0.20000E-07

DISTANCE	CI	ENERGY GROUP	TRA(IN/OUT)	TRA(OUT/IN)	TRA(SM)	FLX(IN/OUT)
1.00	0	0.4000E 06-0.8000E 06	0.	0.40000E 05	0.40000E 05	0.
1.00	0	0.8000E 06-0.1640E 07	0.	0.22000E 06	0.22000E 06	0.
1.00	0	0.1640E 07-0.2150E 07	0.	0.60000E 05	0.60000E 05	0.
1.00	0	0. -0.2150E 07	0.	0.32000E 06	0.32000E 06	0.
3.54	0	0.4000E 06-0.8000E 06	0.80000E 05	0.	0.80000E 05	0.15747E 06
3.54	0	0.8000E 06-0.1640E 07	0.20000E 06	0.	0.20000E 06	0.66431E 06
3.54	0	0.1640E 07-0.2150E 07	0.12000E 06	0.	0.12000E 06	0.88891E 06
3.54	0	0. -0.2150E 07	0.40000E 06	0.	0.40000E 06	0.17107E 07

TIME INTERVAL = 0.20000E-07 -0.30000E-07

DISTANCE	CI	ENERGY GROUP	TRA(IN/OUT)	TRA(OUT/IN)	TRA(SM)	FLX(IN/OUT)
3.54	0	0.4000E 06-0.8000E 06	0.20000E 05	0.	0.20000E 05	0.70801E 05
3.54	0	0.8000E 06-0.1640E 07	0.20000E 05	0.	0.20000E 05	0.24621E 05
3.54	0	0. -0.2150E 07	0.40000E 05	0.	0.40000E 05	0.95422E 05

1)

FLX(OUT/IN)	FLX(SM)
0.	0.83632E 05
0.	0.83998E 07
0.	0.10129E 09
0.	0.10977E 09
0.12343E 06	0.12343E 06
0.97293E 06	0.97293E 06
0.48060E 06	0.48060E 06
0.15720E 07	0.15720E 07
0.	0.15747E 06
0.	0.66431E 06
0.	0.88891E 06
0.	0.17107E 07
0.36837E 05	0.36837E 05
0.15376E 08	0.15376E 08
0.20888E 08	0.20888E 08
0.36301E 08	0.36301E 08

TIMOC (TTS, BY H.KSCHWENDT, EURATOM ISPRA) ** TIME DEPENDENT NEUTRON DECAY IN A C12-SLAB* SAMPLE PRO.NO.3

TIME INTERVAL = 0. -1.00000E-08

DISTANCE PER UNIT	CI	ENERGY GROUP	TRA(IN/OUT)	TRA(OUT/IN)	TRA(SM)	FLX(IN/OUT)
ENERGY						
1.00	0	0.4000E 06-0.8000E 06	0.	0.50000E-01	0.50000E-01	0.
1.00	0	0.8000E 06-0.1640E 07	0.	0.10500E 02	0.10500E 02	0.
1.00	0	0.1640E 07-0.2150E 07	0.	0.10039E 02	0.10039E 02	0.
3.54	0	0.4000E 06-0.8000E 06	0.15000E-00	0.	0.15000E-00	0.20908E-00
3.54	0	0.8000E 06-0.1640E 07	0.39524E 01	0.	0.39524E 01	0.99998E 01
3.54	0	0.1640E 07-0.2150E 07	0.16059E 03	0.	0.16059E 03	0.19860E 03

TIME INTERVAL = 1.00000E-08 -0.20000E-07

DISTANCE PER UNIT	CI	ENERGY GROUP	TRA(IN/OUT)	TRA(OUT/IN)	TRA(SM)	FLX(IN/OUT)
ENERGY						
1.00	0	0.4000E 06-0.8000E 06	0.	1.00000E-01	1.00000E-01	0.
1.00	0	0.8000E 06-0.1640E 07	0.	0.26190E-00	0.26190E-00	0.
1.00	0	0.1640E 07-0.2150E 07	0.	0.11765E-00	0.11765E-00	0.
3.54	0	0.4000E 06-0.8000E 06	0.20000E-00	0.	0.20000E-00	0.39367E-00
3.54	0	0.8000E 06-0.1640E 07	0.23810E-00	0.	0.23810E-00	0.79085E 00
3.54	0	0.1640E 07-0.2150E 07	0.23529E-00	0.	0.23529E-00	0.17430E 01

TIME INTERVAL = 0.20000E-07 -0.30000E-07

DISTANCE PER UNIT	CI	ENERGY GROUP	TRA(IN/OUT)	TRA(OUT/IN)	TRA(SM)	FLX(IN/OUT)
ENERGY						
3.54	0	0.4000E 06-0.8000E 06	0.50000E-01	0.	0.50000E-01	0.17700E-00
3.54	0	0.8000E 06-0.1640E 07	0.23810E-01	0.	0.23810E-01	0.29311E-01

1)

FLX(OUT/IN)	FLX(SM)
0.	0.
0.	0.
0.	0.
0.92092E-01	0.92092E-01
0.18305E 02	0.18305E 02
0.40957E 02	0.40957E 02
0.	0.
0.	0.
0.	0.
0.30858E-00	0.30858E-00
0.11582E 01	0.11582E 01
0.94235E 00	0.94235E 00
0.	0.
0.	0.
0.	0.
0.20908E-00	0.20908E-00
0.99998E 01	0.99998E 01
0.19860E 03	0.19860E 03

1)

FLX(OUT/IN)	FLX(SM)
0.	0.17700E-00
0.	0.29311E-01
0.	
0.	0.39367E-00
0.	0.79085E 00
0.	0.17430E 01
0.	0.30858E-00
0.	0.11582E 01
0.	0.94235E 00
0.	0.20908E-00
0.	0.99998E 01
0.	0.19860E 03

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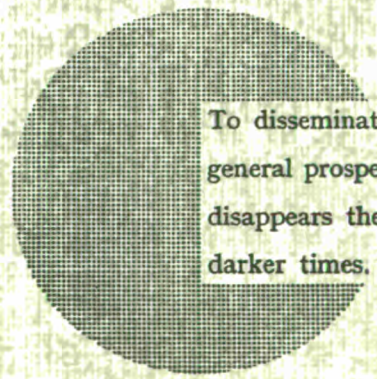

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Alfred Nobel

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